Description of Permit Modification Requested

1. On December 27, 1996, the National Science Foundation issued a permit (97WM–4) to Dr. Rennie S. Holt at the National Oceanic and Atmospheric Administration's (NOAA) Antarctic Marine Living Resources (AMLR) Program after posting a notice in the November 21, 1996 **Federal Register.** Public comments were not received. The issued permit was for the use and release of designated pollutants associated with the construction and operation of a research field camp at Camp Shirreff, Livingston Island, Antarctica (62°28′S60°47′W). During the first season at Cape Shirreff, only limited research activities were conducted as most of the effort was focused on camp construction. In the coming seasons, the AMLR Program proposes to expand research activities, providing a more comprehensive research program. One project of this expanded program proposes to use the doubly labeled water (tritiated and oxygen-18) method to measure the free-ranging foraging energetics of Antarctic fur seals (*Arctocephalus gazella*). Use of tritium labeled water was not included in the original permit request. The scope of this application for a permit modification pertains to waste management issues involved with the use and handling of the radioactive isotope tritium. The duration of the requested modification is coincident with the current permit which expires on April 30, 2001.

All radioisotope materials will be handled only by researchers trained in their proper handling and use. For each season it is anticipated that approximately 55 mCi ^3H_2O will be used for research purposes. All wastes generated from the research activities will be double bagged, packaged in appropriate containers lined with absorbent pads, and will be returned to the University of California Environmental Health and Safety Office, Santa Cruz for disposal. Conditions of the permit modification would include an annual report of all activities involving the tritium and a declaration by the institutional radiation safety officer that all materials returned from the Antarctic have been received.

Joyce A. Jatko,

Acting Permit Officer.

[FR Doc. 97–34038 Filed 12–30–97; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Pub. L. 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 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 December 6, 1997, through December 18, 1997. The last biweekly notice was published on December 17, 1997 (62 FR 66133).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of 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 January 30, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board

Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Carolina Power & Light Company, et al.

[Docket Nos. 50–325 and 50–324]

Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: November 26, 1997.

Description of amendments request: Carolina Power & Light Company (CP&L) has proposed amendments to the Technical Specifications (TS) for the Brunswick Steam Electric Plant Units 1 and 2 (BSEP 1 & 2) to revise certain instrumentation allowable values. The revised values were calculated using a methodology and format consistent with that provided in NUREG–1433, Revision 1, ''Standard Technical Specifications General Electric Plants, BWR/4.'' The current TS are based on the uncertainty associated with the trip unit portion of the instrumentation circuitry. The proposed values are based on the uncertainty associated with the entire instrumentation loop (sensor and trip unit). The NRC has previously approved this methodology for BSEP 1 & 2 as part of a 5 percent power uprate amendment dated November 1, 1996.

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

The proposed changes affect accident mitigation instrumentation allowable values. The changes will not affect the accident mitigation instrumentation functions. No changes will occur in the way in which equipment is operated. Therefore, the probability of a previously evaluated accident can not be affected.

The proposed changes establish the allowable values for certain functions in accordance with the CP&L setpoint methodology, which has been approved, by the NRC, for use at the BSEP. The proposed changes do not affect the actual instrument setpoints. The proposed allowable values were calculated by applying calibration based errors to the trip setpoint values; thereby establishing an operability limit associated with the entire loop of an instrumentation function to ensure sufficient margin to protect analytical limits. The changes do not affect the analytical limits associated with the involved instrumentation functions. The involved instrumentation will continue to perform its accident mitigation functions as designed. Therefore, the consequences of a previously evaluated accident are not increased.

2. The proposed amendments would not create the possibility of a new or different

kind of accident from any accident previously evaluated.

The proposed changes do not affect the actual instrument setpoints nor do they affect the accident mitigation instrumentation functions. No changes will occur in the way in which equipment is operated. The involved instrumentation will continue to perform its accident mitigation functions as designed. Therefore, the proposed license amendments can not create the possibility of a new or different kind of accident.

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

The proposed changes affect accident mitigation instrumentation allowable values. The changes will not affect the accident mitigation instrumentation functions. No changes will occur in the way in which equipment is operated. The proposed changes establish the allowable values for certain functions in accordance with the CP&L setpoint methodology which has been approved, by the NRC, for use at the BSEP. The proposed allowable values were calculated by applying calibration based errors to the trip setpoint values; thereby establishing an operability limit associated with the entire loop of an instrumentation function to ensure sufficient margin to protect analytical limits. The changes do not affect the analytical limits associated with the involved instrumentation functions. The involved instrumentation will continue to perform its accident mitigation functions as designed. Therefore, the proposed license amendments 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: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

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: James E. Lyons.

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: October 29, 1997.

Description of amendment request: Technical Specifications (TS) 3.8.1.1.a.3, 3.8.1.1.b.4, and 3.8.1.1.d.2 presently require a plant shutdown and declaring the redundant required feature inoperable, when the required feature powered from the operable A.C. source is inoperable. The proposed change clarifies the intent of this TS to permit the applicable redundant required feature TS to direct a plant shutdown when required. The proposed amendment changes the existing TS 3.8.1.1.a.3, 3.8.1.1.b.4, and 3.8.1.1.d.2 to eliminate the separate requirement for plant shutdown and instead allows the applicable required redundant feature TS to direct the plant shutdown when required.

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 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 proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS. The changes are consistent with NUREG–1431 and the Commission's Final Policy Statement on Technical Specification improvements.

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

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

The proposed amendment will not introduce any new equipment or require existing equipment to function different from that previously evaluated in the Final Safety Analysis Report (FSAR) or TS. The changes are consistent with NUREG–1431 and the Commission's Final Policy Statement on Technical Specification improvements. The proposed amendment will not create any new accident scenarios, because the change does not introduce any new single failures, adverse equipment or material interactions, or release paths.

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.

Margin of safety for acceptable TS action times have been determined for each TS related system. The proposed change will not alter individual system TS action times. HNP [the Harris Nuclear Plant] proposes to change the requirement to shutdown after expiration of the completion time of an inoperable A.C. source concurrent with an inoperable required feature. Instead of requiring a shutdown, the required feature on the inoperable A.C. source will be declared inoperable and the individual TS will be implemented.

In most cases with both redundant features inoperable, a plant shutdown will be required by TS 3.0.3. In the few instances where additional time is allowed by the individual TS for both redundant required features being inoperable, then an immediate plant shutdown would not be required. The allowed out of service time for loss of individual safety functions has been previously analyzed for HNP TS and NUREG–1431, Revision 1.

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: 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: James E. Lyons.

Florida Power and Light Company, et al.

[Docket Nos. 50–335 and 50–389]

St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: December 1, 1997.

Description of amendment request: The proposed amendment revises the Unit 1 and Unit 2 Environmental Protection Plans (EPP) Section 4, ''Environmental Conditions,'' and Section 5, ''Administrative Procedures,'' to incorporate the proposed terms and conditions of the Incidental Take Statement included in the Biological Opinion issued by the National Marine Fisheries Service (NMFS) on February 7, 1997. The proposed amendment also revises the wording in the Unit 1 EPP to make it consistent with the Unit 2 EPP.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the

probability or consequences of an accident previously evaluated.

The changes are administrative in nature and would in no way affect the initial conditions, assumptions, or conclusions of the St. Lucie Unit 1 or Unit 2, accident analyses. In addition, the proposed changes would not affect the operation or performance of any equipment assumed in the accident analyses.

Based on the above information, we conclude that the proposed changes would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The changes are administrative in nature and would in no way impact or alter the configuration or operation of the facilities and would create no new modes of operation. We conclude that the proposed changes would not create the possibility of a new or different kind of accident.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

As indicated in the discussion of Criterion 1, the changes are administrative in nature and would in no way affect plant or equipment operation or the accident analysis. We conclude that the proposed changes would 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 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: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981–5596.

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Project Director: Frederick J. Hebdon.

IES Utilities Inc.

[Docket No. 50–331]

Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: October 30, 1996.

Description of amendment request: The proposed amendment, included as part of the proposed conversion from current Technical Specifications (CTS) to improved Technical Specifications (ITS), would modify the Surveillance Requirements (SRs) recommended in NUREG–1433 LOC 3.5.1 by revising the combinations (Conditions C, D, G, and I of ITS 3.5.1) of emergency core cooling systems/subsystems that may be out of service. The combinations are supported by the Duane Arnold Energy Center (DAEC) Loss-of-Coolant Accident (LOCA) analysis.

Condition C

ITS 3.5.1 Action C establishes Required Actions and Completion Times for the situation when one core spray (CS) subsystem and one or two residual heat removal (RHR) pump(s) are inoperable. The proposed specification is less restrictive than CTS 3.5.A.4, which allows one RHR pump to be inoperable for 30 days, and CTS 3.5.A.5, which allows two RHR pumps (i.e., the low pressure coolant injection (LPCI) subsystem) to be inoperable for up to 7 days, provided the remaining RHR (i.e., LPCI) active components, both CS subsystems, the containment spray subsystem, and the diesel generators are verified to be operable. The CTS does not allow one CS subsystem *and* one or two RHR pump(s) to be inoperable at the same time. The LOCA analysis presented in NEDC–31310P, (Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis), indicates that an adequate level of protection is provided by the remaining operable ECCS subsystems. The accident analysis also demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. However, another single failure may place the plant in a condition where adequate core cooling may not be available during a DBA–LOCA. Therefore, a Completion Time of 72 hours has been proposed to either restore the inoperable CS subsystem or the inoperable RHR pump(s).

Condition D

ITS 3.5.1 Action D establishes Required Actions and Completion Times for the situation when two CS subsystems are inoperable. The proposed specification is less restrictive than CTS 3.5.A.2, which allows only one CS subsystem to be inoperable. CTS 3.5.A.6 would require the plant to be in Hot Shutdown within 12 hours and Cold Shutdown within the following 24 hours if both CS subsystems were inoperable. With two CS subsystems inoperable, the LOCA analysis presented in NEDC–31310P, (Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis), indicates that the remaining operable low pressure ECCS subsystem consisting of LPCI with four RHR pumps operable (only 3 pumps required), provides adequate protection. However, another single failure may place the plant in a condition where adequate core cooling may not be available during a Design Basis Accident LOCA. Therefore, a Completion Time of 72 hours has been proposed to restore one CS subsystem to operable status.

Condition G

ITS 3.5.1 Action G establishes Required Actions and Completion Times for the situation when HPCI and one RHR pump are inoperable. The proposed specification is less restrictive than CTS 3.5.D.2, which allows continued operation if HPCI is inoperable only if both CSs, LPCI, ADS, and RCIC are verified to be operable. While the LPCI subsystem is technically operable with only 3 of 4 RHR pumps operable, the CTS is currently interpreted by DAEC to require all 4 RHR pumps to be operable for the requirements of CTS 3.5.D.2 to be met, as a single RHR pump has more makeup capability than the HPCI System. Thus for mitigating small and intermediate break LOCAs, one LPCI pump, in combination with ADS, is more than adequate core cooling. The condition of when HPCI and one RHR pump are inoperable is bounded by the analysis in NEDC–31310P, Duane Arnold Energy Center, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis. Since the remaining operable low pressure ECCS subsystems are more than capable of performing their intended function, and RCIC and ADS are Operable, the proposed Action G maintains LOCA analysis assumptions for ECCS Operability. The proposed ITS condition allows 7 days to restore the HPCI System or the RHR pump to operable status. The licensee considers the 7 day Completion Time reasonable in that the LOCA analysis demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. The 7 day Completion Time also provides the benefit of potentially avoiding an unnecessary plant shutdown while the safety functions are still capable of being performed.

Condition I

ITS 3.5.1 Action I establishes Required Actions and Completion Times for the situation when HPCI and one ADS valve are inoperable. The proposed Specification is less restrictive than CTS 3.5.D.2, which allows continued operation if HPCI is inoperable only if both CSs, LPCI, ADS, and RCIC are verified to be operable. While ADS is capable of performing its design function with only 3 of 4 valves operable, per NEDC–31310P, Duane Arnold Energy Center, SAFER/GESTR-

LOCA Loss-of-Coolant Accident Analysis, the CTS requires all 4 ADS valves to be operable for the requirements of CTS 3.5.D.2 to be met. The proposed specification is less restrictive than CTS 3.5.F.2, which allows continued operation when one ADS valve is inoperable only if HPCI is verified to be operable. Since all low pressure ECCS subsystems remain capable of performing their design function and ADS is still capable of performing its design function, ITS 3.5.1 Action I maintains LOCA assumptions to ensure an adequate level of protection is maintained. The proposed condition allows 72 hours to restore the HPCI system or the ADS valve to operable status, since another single failure (i.e., loss of another ADS valve), may place the plant in a condition where adequate core cooling may not be available during a small or intermediate break LOCA.

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:

For Condition C

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

The proposed change will allow one Core Spray subsystem and one or two RHR pump(s) to be inoperable for up to 72 hours. The ECCS subsystems affected by this change are not assumed to be initiators of analyzed events. Therefore, the proposed change does not increase the probability of any accident. The role of these ECCS subsystems is in the mitigation of accident consequences. The proposed change does not allow unlimited continuous operation with the plant in a condition where an additional single failure could result in a loss of ECCS function. The proposed change does not increase the consequences of an accident because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, indicates that an adequate level of protection is maintained by the ADS System and the remaining Operable ECCS subsystems when one Core Spray subsystem and one or two RHR pump(s) are inoperable. Therefore, this change will not involve a significant increase in the 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 will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested or inspected. The change ensures the remaining ECCS capability is adequate to mitigate the consequences of accidents. Therefore, this change will 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 a margin of safety?

The proposed change does not significantly reduce the margin of safety because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, indicates that the plant is protected by the ADS System and the remaining ECCS subsystems when one Core Spray subsystem and one or two RHR pump(s) are inoperable. The accident analysis demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. However, with one Core Spray subsystem and one or two RHR pump(s) inoperable, another single failure may place the plant in a condition where adequate core cooling may not be available during a DBA–LOCA. Therefore, a Completion Time of 72 hours has been assigned to either restore the inoperable Core Spray subsystem or the RHR pump. In addition, this change provides the benefit of potentially avoiding an unnecessary plant shutdown (due to a Completion Time being provided for one Core Spray subsystem and one or two RHR pump(s)) when the remaining ECCS subsystems and the ADS are capable of mitigating potential events. Therefore, this change does not involve a significant reduction in a martin safety.

For Condition D

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

The proposed change will allow both Core Spray subsystems to be inoperable for up to 72 hours. The ECCS subsystems affected by this change are not assumed to be initiators of analyzed events. Therefore, the proposed change does not increase the probability of any accident. The role of these ECCS subsystems is in the mitigation of accident consequences. The proposed change does not allow unlimited continuous operation with the plant in a condition where an additional single failure could result in a loss of ECCS function. The proposed change does not increase the consequences of an accident because accident analysis presented in NEDC–3131OP, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, indicates that an adequate level of protection is maintained by the ADS System and remaining Operable ECCS subsystem when two Core Spray subsystems or inoperable. Therefore, this change will 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 will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The change ensures the remaining ECCS capability is adequate to mitigate the consequences of accidents. Therefore, this change will 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 a margin of safety?

The proposed change does not significantly reduce the margin of safety because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, indicates that the plant is protected by the ADS System and the remaining ECCS subsystem when two Core Spray subsystems are inoperable. The accident analysis demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. However, with both Core Spray subsystems inoperable, another single failure may place the plant in a condition where adequate core cooling may not be available during a DBA–LOCA. Therefore, a Completion Time of 72 hours has been assigned to restore one inoperable Core Spray subsystem. In addition this change provides the benefit of potentially avoiding an unnecessary plant shutdown (due to a Completion Time being provided for both Core Spray subsystems inoperable) when the remaining ECCS subsystem and the ADS are capable of mitigating potential events. Therefore, this change does not involve a significant reduction in a margin of safety.

Condition G

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

The proposed change will allow the HPCI System and one RHR pump to be inoperable for up to 7 days. The ECCS subsystems affected by this change are not assumed to be initiators of analyzed events. Therefore, the proposed change does not increase the probability of any accident. The role of these ECCS subsystems is in the mitigation of accident consequences. The proposed change does not allow unlimited continuous operation with the plant in a condition where an additional single failure could result in a loss of ECCS function. The proposed change does not increase the consequences of an accident because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTRA–LOCA Loss-of-Coolant Accident Analysis, indicated that an adequate level of protection is maintained by the ADS System and the remaining Operable ECCS subsystems when HPCI and one RHR pump are inoperable. Therefore, this change will not involve a significant increase in the 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 will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The change ensures the remaining ECCS capability is adequate to mitigate the consequences of accidents. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluate.

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

The proposed change does not significantly reduce the margin of safety because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR–LOCA Loss-of-Coolant Accident Analysis, indicates that the plant is protected by the ADS System and the remaining ECCS subsystems when HPCI and one RHR pump are inoperable. The accident analysis demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. However, with both HPCI and one RHR pump inoperable, another single failure may place the plant in a condition where adequate core cooling may not be available during an accident. Therefore, a Completion Time of 7 days has been assigned to either restore the inoperable HPCI System or the RHR pump. In addition, this change provides the benefit of potentially avoiding an unnecessary plant shutdown (due to a Completion Time being provided for the HPCI System and one RHR pump inoperable) when the remaining ECCS subsystems and the ADS are capable of mitigating potential events. Therefore, this change does not involve a significant reduction in a margin of safety.

Condtion I

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

The proposed change will allow the HPCI system and one ADS valve to be inoperable for up to 72 hours. The ECCS subsystems affected by this change are not assumed to be initiators or analyzed events. Therefore, the proposed change does not increase the probability of any accident. The role of these ECCS subsystems is in the mitigation of accident consequences. The proposed change does not allow unlimited continuous operation with the plant in a condition where an additional single failure could result in a loss of ECCS function. The proposed change does not increase the consequences of an accident because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTER–LOCA Loss-of-Coolant Accident Analysis, indicates that an adequate level of protection is maintained by the remaining ADS valves (the ADS design function is maintained) in combination with the remaining Operable ECCS subsystems when HPCI and one ADS valve are inoperable. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change will not involve any physical changes to plant systems, structures, or components (SSCs) or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The change ensures the remaining ECCS capability in adequate to mitigate the consequences of accidents. Therefore, this change will 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 a margin of safety?

The proposed change does not significantly reduce the margin of safety because accident analysis presented in NEDC–31310P, Duane Arnold Energy Center SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, indicates that the plant is protected by the remaining ADS valves and the low pressure ECCS subsystems when HPCI and one ADS valve are inoperable. The accident analysis demonstrates that in this condition, the peak clad temperature remains below the regulatory limit. However, with both HPCI and one ADS valve inoperable, another single failure (i.e., of an ADS valve) may place the plant in a condition where adequate core cooling may not be available during a small or intermediate break LOCA. Therefore, a Completion Time of 72 hours has been assigned to either restore the inoperable HPCI System or the ADS valve. In addition, this change provides the benefit of potentially avoiding an unnecessary plant shutdown (due to a Completion Time being provided for the HPCI System and one ADS valve inoperable) when the remaining ECCS subsystems and ADS valves are capable of mitigating potential events. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

Acting NRC Project Director: Richard P. Savio.

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: August 1, 1997 (AEP:NRC:0906H).

Description of amendment requests: The proposed amendments would revise Technical Specification surveillance 4.7.1.2.b. to delete the requirement that the test be performed at a specified secondary steam supply pressure.

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

Criterion 1

The proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

This is an administrative change intended to clarify the technical specification. There will be no change to the test procedure as a result of this clarification. The proposed change better correlates with the accident requirements for which TDAFP [turbine driven auxiliary feed pump] flow is required, and the change is consistent with the present requirement of testing the TDAFP at a secondary side pressure greater than 310 psig.

Criterion 2

The proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not physically modify the plant, nor does it result in the installation of equipment which could introduce a new failure mechanism.

Criterion 3

The proposed change does not involve a significant reduction in a margin of safety. The proposed change does not affect the performance of the TDAFP. Thus, the TDAFP remains capable of providing the required flow under accident conditions, and no safety margins are reduced.

This is an administrative change intended to clarify the technical specification. There will be no change to the test procedure as a result of this clarification

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: Richard P. Savio, Acting.

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: August 11, 1997 (AEP:NRC:1265).

Description of amendment requests: The proposed amendments would revise the Technical Specifications (TS) to allow the filling of the emergency core cooling system (ECCS) accumulators without declaring ECCS equipment 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:

Criterion 1

This amendment request does not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed changes to the T/S represent the possibility of an event that has such a low probability as to not be considered credible. A calculation was performed that demonstrated the CDF resulting from the accumulator fill line operation with all of the conditions assumed above is approximately 3 x 10^{-10} per year. This is well below the NEI guidelines of 1 x 10^{-6} for acceptable risk for a given evolution. Therefore, based on probabilistic considerations and the robust design of the pumps, we conclude the risk associated with this proposed change will not result in a significant increase in the probability or consequences of a previously evaluated accident.

Criterion 2

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The change does not involve a physical change to the plant, but does involve a change in the plant operating configuration. The possibility of a LBLOCA [large break loss of coolant accident] occurring during the accumulation fill evolution has been evaluated and determined to not be credible. Westinghouse has confirmed the accumulator fill line was not modeled in the accident analyses due to the extremely short duration of the fill operation and the extremely small amount of flow that the fill line is capable of passing. The overall effect this configuration would have on the capability of the SI [safety injection] pump to perform its design function, should a LBLOCA occur during the extremely brief window of opportunity, is negligible and would not create a new type of accident.

Criterion 3

This proposed change does not involve a significant reduction in a margin of safety, as the risk from the postulated sequence of events is insignificant. Additionally, engineering evaluation has determined that the real response of an SI pump under the postulated conditions would not be severe. The rugged construction of the pumps, and the design margin built into them, are factors that support the engineering judgment that the affected pump would continue to operate for some time, at some capacity beyond the manufacturer's design limit. As a result of exceeding the limit, the pump may experience some cavitation and require additional corrective maintenance, but would be expected to deliver a significant fraction of its design flow.

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: Richard P. Savio, Acting.

Niagara Mohawk Power Corporation

[Docket No. 50–410]

Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: October 7, 1997.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) to change the setpoints of Surveillance Requirements (SRs) 4.9.6.a, 4.9.6.f, and 4.9.6.g for the refueling platform main hoist. Specifically, each refueling platform crane or hoist used for handling control rods or fuel assemblies within the reactor pressure vessel would be demonstrated operable by:

a. Demonstrating operation of the overload cutoff on the main hoist when the load exceeds 1600 +100/−0 pounds (rather than 1200 +50/−50 pounds).

f. Demonstrating operation of the loaded interlock on the main hoist when the load exceeds 700 +50/−0 pounds (rather than 485 +50/−50 pounds).

g. Demonstrating operation of the redundant loaded interlock on the main hoist when the load exceeds 700 +50/−0 pounds (rather than 550 +50/−50 pounds).

The proposed amendment, in effect, would authorize replacement of the existing triangular refueling platform mast with a round, heavier mast (General Electric Model NF–500) which includes an installed camera/TV 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:

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

The proposed change revises the setpoints for three TS SRs based on modifications to the refueling platform mast. The new mast is essentially a direct replacement for the existing mast, with the exception that the new mast is approximately 400 lbs. heavier, which directly affects the setpoints. No change in the frequency or manner in which the surveillances are performed is proposed. Refueling interlocks will continue to function as designed. No changes to the methods in which plant systems are operated are required. The same design criteria and standards were applied to the new mast, including the seismic capability of the refueling platform with the heavier mast. Therefore, none of the precursors of previously evaluated accidents are affected, and no new failure modes are introduced.

Based on the additional weight of the new mast and camera/TV system, the revised GESTAR [General Electric GESTAR II document NEDE–24011-P-A–11-U5] criteria for fuel rod damage (more conservative threshold level), the use of GE11 [9x9] fuel for the bundle drop analysis, the number of damaged fuel rods has increased slightly for the potential fuel handling accident. The results of this increase were evaluated and dispositioned against the bounding calculation to show that the current USAR [updated safety analysis report] analysis bounds the revised radiological consequences which remain well within the GDC [General Design Criterion] 19 and 10CFR[part]100 limits. The systems that are available to mitigate the consequences of any accident have not been affected and are still capable of performing their required functions. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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 accident previously evaluated.

The proposed change revises the setpoints for three TS SRs based on installation of a new refueling platform which is heavier than the current mast. No change in the frequency or manner in which the surveillances are performed has occurred. Refueling interlocks will continue to function as designed. No changes to the methods in which plant systems are operated are required. The same design criteria and standards were applied to the new mast, including the seismic capability of the refueling platform with the heavier mast. The basic function and operation of the refueling platform is unchanged. The uptravel stop and downtravel mechanical cutoff setpoints are not being changed and will continue to ensure that adequate water shielding is maintained. As such, the change does not introduce any new failure modes or conditions that may create a new or different kind of accident. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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 change revises three TS SR setpoints based on installation of a new refueling platform mast. No change in the frequency or manner in which the surveillances are performed has occurred. Refueling interlocks will continue to function as designed. No changes to the methods in which plant systems are operated are required. The same design criteria and standards were applied to the new mast, including the seismic capability of the refueling platform with the heavier mast. The addition of a camera/TV system will provide enhanced visibility for fuel handling activities and additional assurance that the grapple is oriented over the correct fuel bundle.

The additional weight of the new mast has been evaluated and the operability requirements as described in the TS and TS Bases are unchanged. The modification and revised setpoints do not change the function of the refueling platform main hoist. The revised setpoints will continue to assure the lifting capacity of the main hoist will not be sufficient to result in damage to core internals or the reactor pressure vessel in the event that they are accidentally engaged.

The necessary systems are still available to mitigate any potential radiological consequences of the increased number of damaged fuel rods. The radiological consequences remain within the bounds of the current safety analysis and well below the GDC 19 and 10CFR[Part]100 limits. Therefore, the change does not involve any significant reduction in a margin of safety.

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

Local Public Document Room location: 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: S. Singh Bajwa.

Niagara Mohawk Power Corporation

[Docket No. 50–410]

Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of amendment request: October 31, 1997.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) to support installation of the General Electric Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitor (PRNM) System. The TS changes apply to Sections 2.2, ''Limiting Safety System Settings''; 3/4.3.1, ''Reactor Protection System Instrumentation'' and its corresponding Bases; and 3/4.3.6, ''Control Rod Block Instrumentation.''

Basis for proposed no significant hazards consideration determination: The NUMAC–PRNM will monitor groups of Local Power Range Monitor (LPRM) signals and, together with the Oscillation Power Range Monitor (OPRM), initiate a reactor scram upon identifying neutron flux oscillations characteristic of a thermal-hydraulic instability. The NUMAC–PRNM will replace the existing Average Power Range Monitor (APRM) System and will ultimately support the activation of the OPRM. The proposed modification is in response to Generic Letter 94–02, ''Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactor.'' Except for minor deviations, the proposed TS changes are consistent with General Electric Licensing Topical Report (LTR), NEDC–32410P–A, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC–PRNM) Retrofit Plus Option III Stability Trip Function,'' which was approved by the NRC staff September 5, 1995. Changes with respect to response time testing requirements would be based on Supplement 1 to NEDC–32410P–A, approved by the NRC staff December 26, 1996.

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

As discussed in NEDC–3241OP–A, the NUMAC–PRNM modification and associated changes to the TS involve systems that are intended to detect the symptoms of certain events or accidents mitigating actions. The worst case failure of the systems involved would be a failure to initiate mitigative actions (i.e., scram or rod block), but no failure can cause an accident and therefore the probability of precursors of any accidents previously evaluated is not increased. The NUMAC–PRNM system performs the same operations as the existing equipment, reduces the need for tedious operator action during normal conditions and allows the operator to focus more on overall plant conditions. Automatic self-test and increased operator information available with the NUMARC–PRNM system is likely to reduce the burden during off-normal conditions as well. The NUMAC–PRNM system is compatible with the environmental conditions at the mounting location (e.g., temperature, humidity, seismic, electromagnetic fields) such that system performance will not be degraded when compared to the system being replaced. Therefore, the proposed change will not result in a significant increase in the probability of any accidents previously evaluated.

The proposed changes to the RPS [reactor protection system] and Control Rod Block instrumentation TSs are necessitated by the NUMAC–PRNM replacement. As discussed in the evaluation, in the 4 APRM channel configuration, any two of the four APRM channels and one 2-out-of-4 voter channel in each RPS trip system are required to function for the APRM safety trip function to be accomplished. Therefore, the proposed TS change requires that 3 of the 4 APRM channels be operable. This assures at least two APRM channels to each of the 2-out-of-4 voter channels are available in the event of a single APRM channel failure and one APRM is bypassed. Also, the proposed TS requires a minimum of two 2-out-of-4 voter channels per RPS trip system (i.e., all four voter channels). This assures that at least one voter channel per trip system is available even in the event of a single voter channel failure. Surveillance testing requirements were revised to take advantage of certain features of the NUMAC–PRNM (digital) replacement of the existing analog APRM system. These advantages included improved accuracy, stability , self-testing, reduced drift, and constant time for digital processing. Testing of the RPS and Control Rod Block instrumentation will continue to be performed as described in the evaluation to assure that the reliability and performance of these systems will not be adversely affected.

The proposed NUMAC–PRNM replacement system has been specifically designed to assure that the system response times meet the current acceptance limits (worst case). As a result, due to statistical variations resulting from the sampling and update cycles, the response time is typically faster than required in order to assure the required response time is always met. The architecture of the NUMAC–PRNM system has reduced segmentation compared to the existing PRM system. Examples of the reduced segmentation are combining previously separate functions, several input channels sharing an input board, and a central loop processor for many channels. The replacement equipment includes up to 5 LPRM inputs on a single module compared to one per module on the current system. Up to 17 LPRM signals are processed through one preprocessor. The recirculation flow signals are processed in the same hardware as the LPRM processing. The net effect of these architectural aspects is that there are some single failures that cause a greater loss of ''sub-functionality'' than in the current system. However, other architectural and functional aspects have an offsetting effect. Redundant power supplies are used so that a single failure of AC power has no effect on the overall NUMAC–PRNM system functions while still resulting in a half scram, as does the current system. Continuous automatic self-test also assures that if a single failure does occur, it is much more likely to be detected immediately. The net effect is that from a total system level, there is no increased risk of loss of critical functionality or reduction in safety margins due to the architecture of the replacement system.

Failure analysis indicates that a software common cause failure is not a significant contributor to the unavailability of the NUMAC–PRNM. However, in spite of that conclusion, means are provided within the system to mitigate the effects of such a failure and alert an operator. Therefore, such a failure, even if it occurred, will not increase the consequences of a previously evaluated accident. To reduce the likelihood of common cause failures of software controlled functions, thorough and careful verification and validation (V&V) activities are performed both for the requirements and the implementing software design. In addition, the software is designed to limit the loading that external systems or equipment can place on the system, thus significantly reducing the risk that some abnormal dynamic condition external to the system can cause an overload. For conservatism, however, despite, these V&V activities, common cause failures of software controlled functions due to residual software design faults are assumed to occur. Both the software and hardware are designed to manage the consequences of such failures. Safety outputs are designed to be fail safe by requiring dynamic update of output modules or data signals, where failure to update the information is detected by simple receiving hardware, which in turn, forces a trip. This aspect covers all but rather complex failures where the hardware or software executes a portion of the overall logic but fails to process some portion of the new information (inputs ''freeze'') or some portion of the logic (outputs ''freeze''). To help reduce the likelihood of complex failures, a watchdog timer is used which is updated by a very simple software routine that in turn monitors the operational cycle time of all tasks in the system. The software design is such that as long as all tasks are updating at the design rate, it is likely that software controlled functions are executing as intended. Conversely, if any task fails too update at the design rate, that is a strong indication of at least some unanticipated condition. If such a condition occurs, its watchdog timer will not be updated, the computer will be restarted, and the outputs will detect an abnormal condition and provide an alarm.

It is very difficult to quantify a software common cause failure rate. Analyses for the current system did consider common cause failures and assessed them to be at a rate of about 0.3 times the random failure rate. The reference analysis uses a field basis for the random rates. The analysis for the replacement design uses conservative estimates for failure rates of equipment that are actually a little higher than those assumed for the current equipment. The methodology being applied concludes that the common mode failure rate for the replacement system is somewhat higher than the current system. However, that is offset by more frequent surveillance tests performed by the self-test that result in an estimated slightly lower unavailability for the NUMAC–PRNM scram function compared to the current PRM system. The USAR, in general, considers the failure rate of the function, not that of sub-components. On that basis, there will not be an increase, due to software common cause failure, in the probability of a malfunction analyzed in the USAR.I21The NUMAC–PRNM human-machine interface design does not introduce an increased burden or constraints on the operators' ability to adequately respond to an accident such that there would be more severe consequential effects. The information available to the operators is the same as with the current system. No actions are required by the operator to obtain information normally used and equivalent to that available with the current equipment. However, the replacement system does provide more direct accessible information regarding the condition of the equipment, including automatic self-test, which can aid the operator in diagnosing unusual situations beyond those defined in the licensing basis.

The replacement system has a significantly lower power requirement and is generally smaller, reducing somewhat the seismic loading on the panels. The equipment qualification also includes EMI [electro magnetic induction] emissions which, combined with the fact that the replacement equipment is mounted in its own cabinet (replaces all of the current equipment), minimized the likelihood of significant impact on other existing equipment.

The replacement equipment makes increased use of qualified optical methods to provide both safety and functional isolation between safety-related and nonsafety-related systems. Where fiber optic methods cannot be used, the isolation provided is comparable to or better than that provided in the current system.

The net electrical and thermal load for the replacement system is less than that for the current system. Accordingly, the replacement system had adequate cabinet cooling and no forced cooling is required.

The replacement system meets or exceeds all applicable requirements for separation, independence and grounding. The use of fiber optic connections between the APRM and RBM [rod block monitor] improves the separation and reduces the dependence of the system on common grounds. However, for noise rejection, the equipment design and manufacturing requirements assure improved grounding of the actual equipment.

No change in wiring or grounding external to the panels containing the replacement equipment is necessary for correct operation of the replacement equipment.

NEDC–3241OP–A, Section 3.2.3, discusses different plant configurations for recirculation flow channels, including the case where plants currently (before implementing the NUMAC PRNM system) have four flow channels. Absence of any discussion in the LTR related to separation for plants originally having four flow channels implies that those plants are expected to meet full separation requirements. The LTR includes a further statement that ''The criterion is to maintain equal or better protection against single failures while allowing bypassing of the APRM channel that processes the flow signal.''

The NMPC [Niagara Mohawk Power Corporation] NUMAC PRNM system has four recirculation low channels, but the flow input circuits for two of the four are not separated from each other outside the PRNM panel. As a result, a single failure that causes both of these flow signals to go high could, depending on the specific value, cause the APRM flow biased trip setpoint in two channels to go to the clamped setpoint. If, at the same time, a third channel is bypassed, the APRM flow-biased trip setpoint for the APRM system could be non-conservative. (NOTE: The flow signals are compared to one another. Should the flow signals not be within specified limits, an alarm and a control rod block would be initiated.)

Despite the fact that two of the four flow input circuits are not separated from each other outside the PRNM panel, the replacement system is judged to be adequate with the current field routing of flow signals and meets the LTR criteria. This conclusion is based on the fact that there is no credible fault in the circuits within the duct, in which the flow signals are routed, that can damage the other circuits. Also, there is no credible external fault that can damage the circuits inside the duct. Therefore, it is concluded that the separation between the two flow input circuits is adequate to meet the system single failure requirements in that no credible single failure will disable the flow inputs to more than one APRM channel. Additionally, there are no reload licensing transient analyses that take credit for the flow-biased simulated thermal power scram setpoint.

The replacement design has been specifically designed to have the same or more conservative ''fail safe'' failure modes as the current system. For example, in the case of a single power bus failure, the current system loses about one half of the LPRM information and an output trip occurs. For the replacement system, that failure still results in an output trip, but no LPRM information is lost. In the current system, a static failure in several areas in the system could result in a ''fail-as-is'' state of the outputs. In the replacement system, dynamic coupling starting in the main processor and going to the final output virtually eliminates ''fail-as-is'' failure modes and replaces them with ''fail tripped'' modes.

The replacement system has the same loss of power failure mode as the current system relative to the trip outputs and for loss of AC [alternating current] power. For loss of DC [direct current] power, the replacement system in most cases continues to operate normally due to redundancy of the power supplies. Therefore, the consequences are no different or improved compared to those considered in the USAR.

Both the current system and the replacement system automatically startup on application of power (or re-application). However, the replacement system may take slightly longer to reach normal operation due to initializing activities. However, no USAR evaluations take credit for rapid start of the PRM. Therefore, the slightly longer startup time from point of power application is bounded by the USAR analysis. Upon application of power, once the system is set up for the specific application, it automatically returns to those settings upon application of power. All such setup parameters are stored in non-volatile memory.

Human-machine interfaces (HMI) failures in the current system could be related to misadjusted settings, incorrect reading of meters, and failure to return the equipment to the normal operating configuration. There are comparable failure modes for some of these in the digital system where an erroneous potentiometer adjustment in the current system is equivalent to an erroneous digital entry in the replacement system. Certain potential ''failure to reconfigure'' errors in the current system have no counterpart in the replacement system because any ''reconfiguration'' is automatically returned to normal by the system. Also, since parameters are available for review at any time, even if an error such as a digital entry error occurs, it is more likely that the error would be almost immediately detected by recognition that the displayed value is not the correct one. Failure analysis of the current system assumes certain rates of human error. The rates for the replacement system will be lower, and hence are bounded by the USAR analysis. The NUMAC–PRNM system has been approved as an acceptable neutron monitoring replacement by the NRC.

Therefore, based on the above discussions, the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

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 accident previously evaluated.

NMPC proposes to replace the existing RPS APRM system with the NUMAC–PRNM system and make associated changes to the RPS and Control Rod Block TS instrumentation sections. As discussed in NEDC–3241OP–A, no new system level failure modes are created with the replacement system. The NUMAC–PRNM modification and associated changes to the TSs involve systems that are intended to detect the symptoms of certain events or accidents and initiate mitigating actions. The worst case failure of the systems involved would be a failure to initiate mitigative actions (i.e., scram or rod block), but no failure can cause an accident. This is unchanged from the current system. The proposed changes do not modify the basic functional requirements of the affected equipment, create any new system interfaces or interactions nor create any new system failure modes or sequence of events that could lead to an accident. The replacement system is more tolerant of degraded power than the current system. Software common cause failures can at most cause the system to fail to perform its safety function. As with system level failures, software failures could fail to initiate actions to mitigate the consequences of an accident, but would not cause one. Surveillance testing will continue to be performed to assure reliability and maintain current performance levels.

The NUMAC–PRNM system is a digital system with software (firmware) control. As such, it has ''central'' processing points and software controlled digital processing where the current system has analog and discrete component processing. The result is that the specific failures of hardware and potentially common cause software are different from the current system. Also, automatic self-test results in some cases in a direct trip as a result of a hardware failure where the current system may have remained ''as is.'' However, when these are evaluated at the system level, there are no new effects. In general, the USAR assumes simplistic failure modes (relays for example) but does not specifically evaluate effects added by the NUMAC–PRNM such as self-test detection and automatic trip or alarm. The effects of software common cause failures are mitigated by hardware design and system architecture. The replacement system is fully qualified to operate in its installed location and will not affect other equipment. The NUMAC–PRNM system has been approved as an acceptable neutron monitoring replacement by the NRC. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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 modification and associated TS changes will not adversely affect the performance characteristics of the RPS and Control Rod Block instrumentation nor will it affect the ability of the subject instrumentation to perform its intended function. As stated in NEDC–3241OP–A, the replacement system has improved channel trip accuracy compared to the current system and meets or exceeds system requirements assumed in setpoint analysis. Also, the channel response time is within acceptable limits, the channel indicated accuracy is improved over the current system, and the replacement system does not cause a plant parameter for any analyzed event to fall outside of acceptable limits. The surveillance testing and frequencies proposed will assure reliability of the RPS and Control Rod Block instrumentation. In addition, the subject equipment was qualified, where appropriate, to assure its intended safety function is performed. Therefore, the proposed changes do not involve reduction in a margin of safety.

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

Local Public Document Room location: 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: S. Singh Bajwa.

Pacific Gas and Electric Company

[Docket Nos. 50–275 and 50–323]

Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: July 30, 1997.

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 to add a limiting condition for operation and surveillance requirements for a residual heat removal (RHR) pump trip on low refueling water storage tank (RWST) level to TS 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation.''

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

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

This change assures the availability of the refueling water storage tank (RWST) low-level trip of the residual heat removal (RHR) pumps by establishing limits on the time that a channel can be out of service to 72 hours and establishing surveillance criteria to verify the operation of the logic. The RHR system is used to respond to loss of coolant accidents (LOCAs) and other (e.g., secondary side) accidents that could result in initiation of a safety injection signal, and is not a precursor to any of these events as evaluated in safety analyses. Under accident conditions the RWST serves as the source of water for the emergency core cooling system (ECCS) pumps and the containment spray pumps. The RWST and the RHR pump trip are accident mitigation components and are not precursors for any accident evaluated in the safety analyses.

The existing Technical Specification (TS) would allow one RWST level indication channel to be inoperable indefinitely, and has an allowed outage time (AOT) for two channels inoperable of up to seven days. Additionally, the existing TS does not apply to the RWST low-level RHR pump trip logic. The new TS provides controls that require that all three RWST low-level trip channels be maintained operable while the plant is in Modes 1 to 4, and provides for an AOT for one channel inoperable for up to 72 hours, if the inoperable channel is placed in the cut-out mode within 6 hours. By placing the inoperable channel in the cut-out mode, the possibility of a channel failure causing an RHR pump failure to start at the onset of an accident is precluded even with a single active failure. This assures that the consequences of an accident are not increased.

The change will have no affect on the probability of a physical failure of an RHR pump because it only ensures the presence of a pump trip signal when required. Therefore, there is no increase in the probability of failure of an RHR train to function as designed. This change will have no affect on the probability of any other ECCS equipment failure as it only affects the presence of a trip signal for the RHR pumps.

The new TS 3.3.2 item would provide controls that require that all three RWST level channels be maintained operable while the plant is in operating Modes 1 to 4 (power operation through hot shutdown). By maintaining the three channels operable, the RHR pump actuation/trip logic operability is assured so that the RHR and RWST can in all cases perform their intended accident mitigation functions following a design basis event as evaluated in the safety analyses.

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

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

The RHR system is used to respond to LOCAs and other (e.g., secondary side) accidents that could result in initiation of a safety injection signal. Under accident conditions the RWST serves as the initial source of water for injection by the RHR and other ECCS pumps, and is the source of water for the containment spray pumps. This change does not affect operation of the systems as it relates to their response to accident conditions. It provides additional assurance that the RHR pump trip logic will operate as designed by establishing administrative controls on the time the system is susceptible to a single failure. No new failure modes have been introduced.

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

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

The relevant margin of safety is based on the RHR pumps starting and then automatically stopping at the correct RWST water level. The new TS 3.3.2 item provides controls that require all three RWST level channels be maintained operable while the plant is in Modes 1 to 4. By maintaining the three channels operable, the capability of the RHR pump actuation/trip logic to survive a single active failure is assured. Therefore, the trip logic operability is assured and the margin is preserved. This change also provides additional assurances that the remaining water in the RWST at the time of switchover is consistent with that assumed in the Final Safety Analysis Report and Safety Evaluation Reports.

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

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

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

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

Pennsylvania Power and Light Company

[Docket Nos. 50–387 and 50–388]

Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: June 25, 1997.

Description of amendment request: The amendments would modify the Susquehanna Steam Electric Station, Units 1 and 2 Technical Specifications to reflect an increase in the secondary containment bypass leakage. Specifically, Section 3.6.1.2 is changed to replace the leakage of 1.2 scf per hour for any one main steam line drain with 25.43 scfh for secondary containment bypass leakage from all sources; Section 3.6.1.2 is changed to include the Main Steam Line Drain, high-pressure coolant injection (HPCI) system drain, and reactor core isolation cooling (RCIC) system drain leakages as part of the 300 scfh leakage requirement; and Section 3/4.6.1.2 is changed to include a discussion which related the secondary containment bypass leakage TS to the radiological dose analyses.

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.

Of the potential accidents described in FSAR [Final Safety Analysis Report] Chapters 6 and 15, only a ''Decrease in Reactor Coolant Inventory'' as described in FSAR Section 15.6.5 is affected by the proposed action. The specific accident of concern is a design basis LOCA [loss-of-coolant accident] concurrent with a LOOP [loss-of-offsite power] which results in RPV [reactor pressure vessel] depressurization and failure to recover RPV level above the FW [feedwater] spargers. For this accident, the current licensing basis offsite and control room dose analyses assume a secondary containment bypass leakage rate of 9 scfh and primary containment water (called ESF [engineered safety function]) leakage of 5 gpm. The current licensing basis analyses do not attribute this leakage to any specific pathway.

The proposed action does not increase the probability of a previously analyzed accident in any way. The condition of concern is the result of an accident and as such does not contribute to the initiation of an accident as analyzed in the FSAR.

Of concern is whether or not the proposed action significantly increases the consequences of an accident as previously evaluated. Calculations of off-site dose assuming SCBL [secondary containment bypass leakage] of 28 scfh, primary containment water leakage of 20 gpm, and crediting suppression pool scrubbing show decreases in thyroid dose, but slight increases in whole body dose when compared with dose calculations performed to support the removal of the MSIV–LCS [main steam isolation valve-leakage control system]. This result is expected because the effect of suppression pool scrubbing is factored into the revised licensing basis analysis. Suppression pool scrubbing is effective in reducing iodine release but has no assumed effect on the removal of noble gases. Since the methodology/assumptions for scrubbing are acceptable to the NRC [Nuclear Regulatory Commission] per the guidance in SRP [Standard Review Plan] Section 6.5.5 and the values for decontamination factors are conservative, the judgment may be made that considerable margin is preserved within the analysis.

Although the whole body dose with SCBL of 28 scfh and water leakage of 20 gpm is increased from the previously approved MSIV-LCS dose analysis, the increase is small (about 1 rem at the two hour site boundary; less than 0.1 rem 30 day LPZ [low population zone]). The total dose including the increase is still well below the 10CFR100 whole body regulatory limit of 25 rem to which SSES [Susquehanna Steam Electric Station] was licensed. No change in operating procedures is anticipated. Calculated post accident control room thyroid dose decreases as a result of this change, and the increase in control room whole body dose is less than 0.05 rem, well below the 10CFR50, Appendix A, GDC [General Design Criterion] 19 dose limits outlined in NUREG–0800. Thus, no appreciable effect on operator response will occur as a result of this change.

The addition of the HPCI and RCIC Steam Line Drains to the Tech Spec for MSIV leakage is being performed as a result of the modification which eliminated the MSIV Leakage Control System (MSIV LCS). At the time this modification was performed, these lines were not identified as potential SCBL pathways. However, because leakage from the HPCI and RCIC drain lines are part of the same pathway to the condenser which is now used by the main steam line drains (MSLD) and included in the Technical Specifications, they must be combined with the MSIV's and MSLD to be less than 300 scfh. This change only affects the accounting of the various drain leakages in the valve testing program. The justification for this change is the same justification provided in the ITS [Improved Technical Specification] submittal (PLA–4488, August 1, 1996) which adds the MSLD to this Technical Specification. The test pressure change to allow testing at Pa was previously proposed in PLA–4502, September 23, 1996. One additional change to delete a footnote related to the removal of the MSIV Leakage Control System is included because this system has been removed from Susquehanna SES.

Since the increase in SCBL and primary containment water leakage result in only a small increase in the doses previously evaluated by the NRC and the other changes do not affect the dose analyses, the proposed change does not result in 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.

Because the FSAR analysis already assumes SCBL and ESF leakage occur and the other changes do not affect the type of accident[s] that are postulated to occur, the proposed change does not present the possibility of an accident of a different type. Additionally, the change in dose analysis methodology does not create an accident or malfunction of a different type since it only involves the analysis of the effects of such accidents or malfunctions.

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

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

This question addresses changes in system parameters only. Dose consequences are addressed in Section 1 above. The only Technical Specification dealing with SCBL is T.S. 3.6.1.2 which requires the leakage from any one Main Steam Line Drain (MSLD) Valve to be less than or equal to 1.2 scfh when tested at Pa (45.0 psig). As noted earlier, the current licensing basis accident dose analysis assumes a total of 9 scfh for bypass leakage and 5 gpm for primary containment water leakage but does not attribute them to any particular source. The proposed action increases the assumed SCBL from 9 to 28 scfh and water leakage from 5 gpm to 20 gpm. These leakage rates are insignificant in terms of SGTS [standby gas treatment system] flows or water loss from ECCS systems. These leakage rates do not affect building temperatures or pressures so that they become closer to acceptance limits. Likewise, no other system parameter values become closer to limits as a result of these changes in leakage. Consequently, the existing margin of safety between the licensing basis analysis and system parameter acceptance limits is not reduced. The changes to the HPCI, RCIC, and main steam line drain leakage only affect the accounting for the various leakages in the leakage testing program. The deletion of the footnote is administrative because the MSIV Leakage Control System has been removed from the Susquehanna SES. The change in test pressure was previously evaluated in PLA–4502, September 23, 1996. Thus, no decrease in margin of safety results.

The NRC staff has reviewed the licensee's analysis and notes that a discussion of the administrative change to delete a footnote in Section 3.6.1.2 is in the third section of the no significant hazards consideration. The staff finds that this administrative change also does not involve a significant increase in the probability or consequences of an accident previously evaluated and does not create the possibility of a new or different kind of accident from any accident previously evaluated. Based on this staff 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: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

NRC Project Director: John F. Stolz.

Pennsylvania Power and Light Company

[Docket No. 50–387]

Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: August 26, 1997.

Description of amendment request: The amendment would modify the Susquehanna Steam Electric Station, Unit 1 Technical Specifications to change the definitions in Section 1.0 to make them applicable to ATRIUM–10 fuel (reflecting the new design), to include the Unit 1 Cycle 11 flow dependent minimum critical power ratio (MCPR) Safety Limits in Sections 2.1.2 and 3.4.1.1.2, to change Section 5.3.1 to reflect the ATRIUM–10 design, and to include Siemens Power Corporation methodology topical reports and references to the methodology in Section 6.9.3.2.

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

The applicable sections of the FSAR [Final Safety Analysis Report] are Chapters 5, 6.3, 9, and 15 of the FSAR. Chapter 5 discusses the results of the ASME [American Society of Mechanical Engineers] overpressure analysis for the reactor pressure boundary. Chapter 6.3 discusses the LOCA [loss-of-coolant accident]. Chapter 9 discusses fuel storage and handling. Chapter 15 describes the transient and accident analyses, a majority of which have been dispositioned to be non-limiting. A discussion of the impact of the Technical Specification changes is provided below.

The change to Definitions 1.2 and 1.3 makes the definitions applicable to ATRIUM™–10. There are no effects on safety functions from this change.

A cycle specific MCPR Safety Limit analysis was performed for PP&L [Pennsylvania Power and Light Company] by SPC [Siemien Power Corporation]. This analysis used NRC [Nuclear Regulatory Commission] approved methods described in Technical Specification Reference 13 (ANF–524(P)(A), Revision 2 and Supplement 1 Revision 2), as modified by EMF–97–010(P), Rev. 1. The SAFETY LIMIT MCPR calculation statistically combines uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and the uncertainty in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The SAFETY LIMIT MCPRs are specified as a function of core flow. The resulting two-loop and single-loop values (Technical Specification Sections 2.1.2 and 3.4.1.1.2) are included in the proposed change. Thus, the cladding integrity and its ability to contain fission products are not adversely affected.

The MCPR methodology for ATRIUM™–10 fuel (SPC report EMF–97–010(P), Rev. 1), included in the revised Technical Specifications via reference (Section 6.9.3.2) and previously approved by the NRC for Unit 2 Cycle 9, describes conservative methods for developing the MCPR Safety Limits and Operating Limits for the U1C11 reload of ATRIUM™–10 fuel in the Susquehanna Steam Electric Station. This methodology conservatively accounts for a flow dependence in the ATRIUM™–10 critical power test data as well as an increased correlation uncertainty for high local peaking factor rods. The results of using this methodology are core flow dependent MCPR Safety Limits plus conservative MCPR Operating Limits for Unit 1 Cycle 11. The resulting MCPR Safety Limits and Operating Limits will continue to assure that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. Thus, the cladding integrity and its ability to contain fission products are not adversely affected. The proposed change in MCPR methodology does not physically affect the plant or its systems.

Using the approach discussed in EMF–97–010(P), Rev. 1, analyses of the Pump Seizure accident with the new MCPR methodology (SPC report EMF–97–010(P), Rev. 1) will demonstrate that the NRC acceptance criterion (i.e., small fraction of 10CFR100 dose limits) is met.

The change to the Design Features (Section 5.3) increases the maximum allowable lattice average enrichment. Analyses have demonstrated that the ATRIUM™–10 fuel will remain subcritical (k-effective < 0.95) in both the spent fuel pool and the new fuel vault. Thus, the change to maximum allowable lattice average enrichment has no impact on safety functions. The description of a fuel assembly (Section 5.3) is also revised to reflect the ATRIUM™–10 central water channel, and reference to an active fuel length of 150 inches was deleted. This change reflects the physical characteristics of the ATRIUM™–10 fuel and has no impact on the probability or consequences of an event.

Included in the revised Technical Specifications via reference (Section 6.9.3.2) are additional NRC approved methodology reports. The NRC approved topical reports contain methodology which is used to assure safe operation of Unit 1 with ATRIUM™–10 fuel. These methodologies assure that the core meets appropriate margins of safety for all expected plant operational conditions ranging from refueling and cold shutdown of the reactor through power operation. Thus, the results obtained from the analyses will provide assurance that the reactor will perform its design safety function during normal operation and design basis events.

The BASES changes for Section 2.1.1 (THERMAL POWER, Low Pressure or Low Flow) reflect that the Safety Limit is valid for both 9x9–2 and ATRIUM™–10. BASES for Section 2.1.2 were changed to refer to Section 6.9.3.2 for applicable references.

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

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

The changes to the Unit 1 Technical Specifications (Definitions, MCPR safety limits, Design Features, and inclusion of methodology references) to allow use of ATRIUM™–10 fuel do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The applicable Technical Specification Sections include 1.0, 2.0, 3/4.4, 5.3, and 6.9.3.2.

The changes to the Unit 1 Technical Specifications discussed in Item 1 above do not require any physical plant modifications, physically affect any plant components, or entail significant changes in plant operation. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The consequences of transients and accidents will remain within the criteria approved by the NRC. The proposed MCPR Safety Limits and the NRC approved methods and revised MCPR methodology detailed in the references added to Section 6.9.3.2 maintain an equivalent margin of safety as defined in the BASES of the applicable Technical Specification sections.

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

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

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

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

NRC Project Director: John F. Stolz.

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 amendment requests: June 18, 1997.

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.8.1, ''AC Sources—Operating'' and applicable Bases. This change will more clearly reflect safety analysis and testing conditions.

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

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

The proposed change would revise Technical Specification (TS) TS 3.8.1, ''AC Sources—Operating,'' Surveillance Requirement (SRs) 3.8.1.1, 3.8.1.2, 3.8.1.7, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, 3.8.1.15, 3.8.1.16, 3.8.1.17, 3.8.1.19, and 3.8.1.20 and applicable Bases to more clearly reflect surveillance test conditions and system design requirements. Changes to the SRs include more restrictive voltage and frequency acceptability limits. The new requirements reflect the system design requirements in order to ensure Class 1E system operability, meet the requirements of the safety analysis, and to agree with the existing test surveillances.

In addition, the discussion regarding design basis reactive power loading is eliminated since this cannot be readily controlled during testing.

Operation of the facility would remain unchanged as a result of the proposed change and no assumptions or results of any accident analyses are affected. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change would revise Technical Specification (TS) TS 3.8.1, ''AC Sources—Operating,'' Surveillance Requirement (SRs) 3.8.1.1, 3.8.1.2, 3.8.1.7, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, 3.8.1.15, 3.8.1.16, 3.8.1.17, 3.8.1.19, and 3.8.1.20 and applicable Bases to more clearly reflect surveillance test conditions and system design requirements.

Operation of the facility would remain unchanged as a result of the proposed change. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would revise Technical Specification (TS) TS 3.8.1, ''AC Sources—Operating,'' Surveillance Requirement (SRs) 3.8.1.1, 3.8.1.2, 3.8.1.7, 3.8.1.10, 3.8.1.11, 3.8.1.12, 3.8.1.13, 3.8.1.14, 3.8.1.15, 3.8.1.16, 3.8.1.17, 3.8.1.19, and 3.8.1.20 and applicable Bases to more clearly reflect surveillance test conditions and system design requirements. Changes to the SRs include more restrictive voltage and frequency acceptability limits. The new requirements reflect the system design requirements in order to ensure Class 1E system operability, meet the requirements of the safety analysis, and to agree with the existing test surveillances.

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 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: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

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 amendment requests: November 14, 1997 (supersedes February 1, 1994, amendment request).

Description of amendment requests: The licensee proposes to revise the licensing basis as described in the Updated Final Safety Analysis Report Section 3.5, ''Missile Protection,'' to allow the use of NUREG–0800, ''Standard Review Plan'' methodology in evaluating tornado-generated missiles. In particular, a probability based criteria is proposed to evaluate missile barrier requirements consistent with Section 3.5.1.4 of NUREG–0800.

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.

NUREG–0800, Standard Review Plan (SRP) Section 3.5.1.4, Revision 0 and Section 3.5.1.5 Revision 1 provide a conservatively acceptable probability threshold for safety due to damage caused by postulated missile strikes. Section 3.5.1.4, Revision 0 uses 10^{-7} per year for a tornado-generated missile strike, and Section 3.5.1.5 Revision 1 uses 10^{-7} per year for exceeding 10 CFR Part 100 limits.

The proposed criteria of probability of damage to critical exposed equipment (as defined in San Onofre Updated Final Safety Analysis Report proposed Table 3.5–13) of 10^{-7} per year per unit is consistent with this guidance.

The probability of damage to exposed critical components due to a postulated missile strike of 10^{-7} is so small as to be negligible. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This amendment request establishes a conservative criteria for tornado-generated missiles consistent with the SRP guidance and will not create a new or different kind of accident from any accident that has been previously evaluated.

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

This proposed change is consistent with the methodology and acceptance criteria of the SRP, and the SRP criteria ensures that there will be no undue risk to the health and safety of the public. Therefore, there will be 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 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: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Attorney for licensee: T.E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

NRC Project Director: William H. Bateman.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia

Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

[Docket Nos. 50–424 and 50–425]

Date of amendments request: August 8, 1997, as supplemented October 10, 1997. This application and supplement supersedes the October 4, 1996, application, noticed in the **Federal Register** on November 19, 1996 (61 FR 58903), in its entirety.

Description of amendments request: The proposed amendments would change the Technical Specifications to credit soluble boron in the spent fuel pool for maintenance of subcriticality and increase the allowable fuel enrichment to 5.0 percent U–235 as follows:

1. Revisions to the Table of Contents

The Table of Contents would be revised to include two additional Technical Specifications 3.7.17, ''Fuel Storage Pool Boron Concentration,'' and 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool'' and add Figures 3.7.18–1, 3.7.18–2, and 4.3.1–1 through 4.3.1–9 describing burnup credit, checkerboard configurations and interface requirements. These changes would be added to support crediting soluble boron in the fuel storage pool criticality analyses.

2. Addition of Technical Specifications 3.7.17 and 3.7.18

Technical Specifications 3.7.17, ''Fuel Storage Pool Boron Concentration,'' and 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool,'' would be added to credit soluble boron in the fuel storage pool criticality analyses, and specify acceptable enrichment-burnup combinations for storage of fuel in the fuel storage pool.

3. Revision to Technical Specification 4.3.1.1

Design Features Section 4.3.1.1 would be revised to reflect the increased maximum enrichment assumed in the fuel storage pool criticality analyses, add a requirement to maintain K_{eff} less than 1.0 when fully flooded with unborated water, change the 0.95 K_{eff} requirement from ''if fully flooded with unborated water'' to ''when fully flooded with water borated to 450 ppm (Unit 1) or 500 ppm (Unit 2),'' and to add a reference to Specification 3.7.18 for allowable enrichment-burnup combinations. Requirements for fuel that do not meet the requirements of Specification 3.7.18, would also be added to Section 4.3.1.1, including Figures 4.3.1–1 through 4.3.1–9 depicting acceptable enrichment-burnup requirements and checkerboard configurations.

4. Revisions to the Table of Contents (Bases)

The Table of Contents would be revised to include two additional Technical Specification Bases Sections B 3.7.17 ''Fuel Storage Pool Boron Concentration'' and B 3.7.18 ''Fuel Assembly Storage in the Fuel Storage Pool.''

5. Addition of Bases for Technical Specifications 3.7.17 and 3.7.18

Two additional Technical Specification Bases Sections B 3.7.17, ''Fuel Storage Pool Boron Concentration'' and B 3.7.18, ''Fuel Assembly Storage in the Fuel Storage Pool'' would be added to credit soluble boron in the fuel storage pool criticality analyses.

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 radiological consequences of 5.0 weight percent U–235 fuel on accidents previously evaluated in the Vogtle FSAR [Final Safety Analysis Report] are not significant. Increasing the enrichment up to and including 5.0 weight percent U–235 has minor effects on the radiological source terms and subsequently the potential releases both normal and accidental are not significantly affected. Evaluations performed (WCAP–12610–P–A, Reference 5 [of the licensee's application]) considered the source term, gap fraction, and the accident doses for a maximum fuel enrichment of 5.0 weight percent U–235. It was concluded that operating with and storing fuel with 5.0 weight percent U-235 enrichment may result in minor changes in the normal annual releases of long half-life fission products that are not significant. Also, the radiological consequences of accidents are minimally affected due to the very small changes in the core inventory and the fact that the currently assumed gap fractions remain bounding.

The use of the slightly higher enrichment for VEGP [Vogtle Electric Generating Plant] fuel will not result in burnups in excess of those currently allowed for VEGP. The cycle design methods and limits will remain the same as are currently licensed. Therefore, the use of fuel with the higher enrichment will not result in conditions outside those currently allowed for VEGP.

There is no increase in the probability of a fuel assembly drop accident in the fuel storage pool when considering the presence of soluble boron in the pool water for criticality control. The handling of the fuel assemblies in the fuel storage pool has always been performed in borated water.

Fuel assembly placement will be controlled pursuant to approved fuel handling procedures and will be in accordance with the spent fuel rack storage configuration limitations in the Technical Specifications. The consequences of a misplaced assembly have been included in the analysis supporting this revision to the Technical Specifications.

There is no increase in the consequences of the accidental misloading of a fuel assembly into the fuel storage pool racks because criticality analyses demonstrate that the pool will remain subcritical following an accidental misloading of an assembly. There are no credible dilution events that reduce the subcriticality margin below the 5% margin recommended in NRC guidance (references 1, 2, and 3 [of the licensee's application]). Even if the fuel storage pool were diluted to a boron concentration of 0 ppm the No Soluble Boron 95/95 analysis demonstrates that the pool will remain subcritical. The proposed Technical Specifications limitations will ensure that an adequate fuel storage pool boron concentration will be maintained.

There is no increase in the probability of the loss of normal cooling to the fuel storage pool water due to the presence of soluble boron in the pool water for subcriticality control, because a concentration of soluble boron similar to the proposed limit has always been maintained in the fuel storage pool water.

The loss of normal cooling to the fuel storage pool will cause an increase in the temperature of the fuel storage pool water. This will cause a decrease in water density which would normally result in an addition of negative reactivity. However, since Boraflex is not considered to be present, and the fuel storage pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. The amount of soluble boron required to offset this postulated accident was evaluated for the allowed storage configurations. The amount of soluble boron necessary to mitigate these accidents and ensure that the K_{eff} will be maintained less than or equal to 0.95 has been included in the fuel storage pool boron concentration. Because adequate soluble boron will be maintained in the pool water, the consequences of a loss of normal cooling to the fuel storage pool will not be increased.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The potential for criticality accidents in the fuel storage pool are not new or different types of concerns. The potential criticality accidents have been reanalyzed in the Criticality Analysis report (Enclosure 5 [of the licensee's application]) to demonstrate that the pool remains subcritical.

Soluble boron has been maintained in the fuel storage pool water since its initial operation. The possibility of a fuel storage pool dilution is not affected by the proposed change to the Technical Specifications. Therefore, the implementation of Technical Specification controls for the soluble boron will not create the possibility of a new or different kind of accidental pool dilution.

With credit for soluble boron now a major factor in controlling subcriticality, an evaluation of fuel storage pool dilution events was completed. The results of the evaluation concluded that no credible events would result in a reduction of the criticality margin below the 5% margin recommended by the NRC. In addition, the No Soluble Boron 95/95 criticality analysis assures that dilution to 0 ppm will not result in criticality.

Proposed Technical Specifications 3.7.17, 3.7.18 and 4.3.1.1 which ensure the maintenance of the fuel storage pool boron concentration and storage configuration, do not represent new concepts. The actual boron concentration in the fuel storage pool has been maintained at a higher value than the proposed limits for the Unit 1 and 2 fuel storage pools for refueling purposes. The criticality analysis (Enclosure 5 [of the licensee's application]) determined that a boron concentration of 450 ppm (Unit 1) and 500 ppm (Unit 2) results in a K_{eff} [less than or equal to] 0.95.

There is no significant change in plant configuration, equipment design, or usage of plant equipment. The safety analysis for dilution accidents has been expanded; however, the criticality analyses assure that the pool will remain subcritical with no credit for soluble boron. Therefore, the proposed changes will not create the possibility of a new or different kind of accident.

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

Proposed Technical Specifications 3.7.17, 3.7.18, and 4.3.1.1 and the associated fuel storage pool boron concentration and storage requirements will provide adequate margin to assure that the fuel storage array will always remain subcritical by the 5% margin recommended by the NRC. Those limits are based on the criticality analysis (Enclosure 5 [of the licensee's application]) performed in accordance with the Westinghouse fuel storage rack criticality analysis methodology described in Reference 4 [of the licensee's application].

While the criticality analysis utilized credit for soluble boron, the storage configurations have been defined using K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron.

Soluble boron credit is used to offset off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the fuel storage pool K_{eff} is maintained less than or equal to 0.95.

The combination of the No Soluble Boron 95/95 K_{eff} calculation which shows that the K_{eff} will remain less than 1.0 when flooded with unborated water and the unavailability of the large volumes of water which are necessary to dilute the fuel storage pool to a K_{eff} of > 0.95, provide a level of safety comparable to the conservative criticality analysis methodology required by References 1, 2, and 3 [of the licensee's application].

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia

Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

[Docket Nos. 50–424 and 50–425]

Date of amendment request: September 4, 1997, as supplemented November 20, 1997.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to change the capacity of the Vogtle Unit 1 spent fuel storage pool from 288 to 1476 assemblies, and would revise the design features description to reflect the criticality analyses and storage cell spacing. Specifically, the changes would be as follows:

1. Figure 3.7.18–1 would be replaced with a revised figure based on the criticality analyses for the Unit 1 racks containing boral.
2. The criticality information for Unit 2 would be placed unchanged into Section 4.3.1.2, and Section 4.3.1.1. would be revised to address Unit 1.
3. Design Features Section 4.3.1.1.c would be revised to indicate 600 ppm as the required amount of soluble born to maintain K_{eff} less than or equal to 0.95.
4. Design Features Section 4.3.1.1.d would be revised to include the reference K_{eff} that is equivalent to the combination of burnup and initial enrichment defined by Figure 3.7.18–1.
5. Design Features Section 4.3.1.1.e would be revised to indicate that fuel assemblies with up to 5 weight percent U–235 may be stored in 3-out-of-4 checkerboard storage configurations; delete Figure 4.3.1–1; eliminate the reference to 2-out-of-4 storage for the Unit 1 pool and include the reference K acceptable for all cell storage in the Unit 1 fuel storage racks.
6. Design Features Section 4.3.1.1.f would be revised to include the pitch of the Unit 1 fuel storage racks.

7. Design Features Section 4.3.3 would be revised to indicate the Unit 1 fuel storage pool capacity of 1476 fuel assemblies.

8. The titles on Figures 4.3.1–4, 4.3.1–6, and 4.3.1–7 would be revised to reflect the elimination of 2-out-of-4 storage configuration requirements for the Unit 1 fuel storage pool.

Changes to the TS Bases are also proposed.

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

1. The analyses methodologies are the same as previously approved for use by the NRC. The results of the analyses resulted in fuel pool boron concentrations, and fuel assembly storage limitations that are similar to those already submitted to the NRC. The increased number of fuel assemblies will remain less than the number previously accepted by the NRC for storage in VEGP [Vogtle Electric Generating Plant] Unit 2, which has a similarly designed and constructed facility, with the exception of the number of fuel storage locations.

Therefore, based on the conclusions of the above analysis, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The effects of accidents that could affect the fuel were analyzed for the fuel storage racks, however the types of accidents have not changed. The fuel to be stored in the Unit 1 pool is expected to meet the all cell storage requirements. The racks will be placed in the Unit 1 pool without lifting any loads over spent fuel. After installation of the new racks, the Unit 1 pool will have 1476 storage locations which is well within the 2098 locations that the pool and structure is capable of storing, based on its similarity to the Unit 2 pool.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident.

3. The changes to the technical specifications are necessary to incorporate the parameters resulting from the criticality analyses. The criticality analyses were performed using methods and criteria previously accepted by the NRC. The requirements are similar to the previously submitted requirements. The margins of safety provided by the previous technical specifications are not significantly affected because the new racks are based on the same acceptance values. The larger number of fuel assemblies to be stored in the Unit 1 pool remains well within the capability of the pool.

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia

Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

[Docket Nos. 50–424 and 50–425]

Date of amendment request: November 20, 1997.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to provide for the following with regard to the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation trip setpoints:

1. The inequalities as they are applied to the Trip Setpoint column of Tables 3.3.1–1 and 3.3.2–1 would be deleted, and the column heading would be changed from ''Trip Setpoint'' to ''Nominal Trip Setpoint.''

2. A footnote would be added to the new ''Nominal Trip Setpoint'' column of Tables 3.3.1–1 and 3.3.2–1 that would allow the trip setpoints to be set more conservative than the nominal value as necessary to respond to plant conditions.

3. The Allowable Value for Table 3.3.1–1, Function 14.b, Turbine Trip—Turbine Stop Valve Closure, would be revised from ''[greater than or equal to] 96.7% open'' to ''[greater than or equal to] 90% open.''

4. Footnotes l and m of Table 3.3.1–1 would be revised to refer to the ''Nominal Trip Setpoint'' and delete the inequalities applied to the trip setpoints.

5. A superscript ''(a)'' would be deleted from the heading of the ''Trip Setpoint'' column on page 6 of 8 of Table 3.3.1–1.

6. Notes 1 and 2 to Table 3.3.1–1, Overtemperature ΔT and Overpower ΔT, respectively, would be revised to refer to the ''Nominal Trip Setpoint.'' In addition, these notes will be revised to delete the inequalities from the values for the constants K_1 through K_6 (except for K_5 [greater than or equal to] 0 for decreasing temperature and $K_6 = 0$ for T [less than or equal to] T''), and for T', T'', and P'.

7. The inequality applied to the ESFAS Allowable Value for Steam Line Pressure—Low (Table 3.3.2–1, Function 1.e) would be changed from ''[less than or equal to]'' to ''[greater than or equal to].''

Associated changes to the TS Bases are also proposed.

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

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

No. The proposed changes affect only the presentation of the trip set points for the RTS and ESFAS in the VEGP [Vogtle Electric Generating Plant] Units 1 and 2 TS. The calibration of the channels whose setpoints are specified in the TS will continue to be performed in a manner consistent with the setpoint methodology described in WCAP–11269 Rev. 1. There will be no adverse effect on the ability of those channels to perform their safety functions as assumed in the safety analyses. Since there will be no adverse affect on the trip setpoints or the instrumentation associated with those trip setpoints, there will be no increase in the probability of any accident previously evaluated. Similarly, since the ability of the instrumentation to perform its safety function is not adversely affected, there will [be] no increase in the consequences of any accident previously evaluated.

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

No. The proposed change affects only the presentation of trip setpoint requirements in the TS. Plant operation will not be changed, and the response of safety related equipment as assumed in the accident analyses will not be adversely affected. Therefore, the proposed change does not involve a new or different kind of accident than any previously evaluated.

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

No. As described above, the RTS and ESFAS instrumentation will remain capable of performing its safety function as assumed in the accident analyses. The treatment of trip setpoints as nominal values is consistent with the methodology used to establish those setpoints. As such, margin is not affected by the proposed change.

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

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

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

NRC Project Director: Herbert N. Berkow.

Vermont Yankee Nuclear Power Corporation

[Docket No. 50–271]

Vermont Yankee Nuclear Power Station, Windham County, Vermont

Date of amendment request: October 10, 1997, as supplemented October 31, 1997.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to reflect the installation of a generator no-load disconnect to facilitate use of the main step-up transformer backfeed as the delayed access offsite power source. Also, the amendment would revise existing limiting conditions for operation and required action statements for operation with inoperable ac power sources to be consistent with current guidance.

Specifically, the changes proposed are: (1) TS Limiting Conditions for Operation Section—Normal Operation, 3.10.A.4 (2) TS Limiting Conditions for Operation Section—Operation with Inoperable Components, 3.10.B.3, (3) TS Surveillance Requirements—Normal Operation, 4.10.A.4, (4) TS Surveillance Requirements—Operation with Inoperable Components, Section 4.10.B.3, (5) Bases Section 3.10.A, (6) Bases Section 3.10.B, (7) Bases Section 4.10.A, and (8) Bases Section 4.10.B

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

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

The proposed amendment removes credit for the Vernon Tie, Vermont Yankee's station blackout source of power, from the Technical Specifications and reflects the installation of the generator no load disconnect as part of the backfeed. Neither the backfeed through the main transformers nor the Vernon Tie are accident initiators; therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The change does not affect the capability, availability, maintenance or operation of the Vernon Tie. Installation of the generator no load disconnect switch is being implemented by a design change in order to enhance plant safety by reducing time necessary to establish the backfeed through the main transformer. A separate 10 CFR 50.59 evaluation is being prepared to document that the modification does not create an unreviewed safety question.

The proposed amendment also clarifies the allowable out of service times, and required actions; and updates surveillance requirements for the immediate and delayed access offsite power sources. These changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Modification of a technical specification out of service time and required action cannot affect the probability or consequences of an accident. Enhancing surveillance requirements to provide assurance that the backfeed can be achieved when required and to provide assurance that remaining power sources are available when an offsite source is unavailable improves plant safety and does not increase the probability or consequences of an accident.

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

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

The proposed amendment removes the Vernon Tie, Vermont Yankee's station blackout source of power, as a delayed access source from the Technical Specifications and reflects the improvements to the main transformer backfeed delayed access source because of installation of the generator no load disconnect. Neither the removal of the Vernon Tie from Technical Specifications nor the improvements to the delayed access power source (backfeed) can create the possibility of a new or different kind of accident from any previously evaluated.

The proposed amendment also clarifies the allowed outage times, and action statements; and updates surveillance requirements for the immediate and delayed access offsite power sources. A clarification of a technical specification out of service time and required action cannot create a new or different kind of accident from any accident previously evaluated. Enhancing surveillance requirements to provide assurance that the backfeed can be achieved when required and to provide assurance that remaining power sources are available when an offsite source is unavailable improves plant safety and cannot create a new or different kind of accident from any accident previously evaluated.

Therefore, this change would not create the possibility of a different type of accident than previously evaluated.

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

The proposed amendment removes the Vernon Tie, Vermont Yankee's station blackout source of power, as a delayed source of offsite power from the Technical Specifications and reflects the improvements to the main transformer backfeed delayed access source because of installation of the generator no load disconnect. No existing safety margins are adversely affected. The backfeed is modified so that it may be established in sufficient time to "assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded". Vernon Tie will not be affected by the modification and remain available as a station blackout source; thus this change does not involve a significant reduction in the margin of safety.

The proposed amendment also clarifies the allowed out of service times, and required actions; and updates surveillance requirements for the immediate and delayed access offsite power sources. A clarification of a technical specification out of service time and required action does not involve a significant reduction in the margin of safety in the Technical Specifications. Enhancing surveillance requirements to provide assurance that the backfeed can be achieved when required and to provide assurance that remaining power sources are available when an offsite source is unavailable improves plant safety and does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

NRC Project Director: Ronald Eaton, Acting Director.

Vermont Yankee Nuclear Power Corporation

[Docket No. 50–271]

Vermont Yankee Nuclear Power Station, Windham County, Vermont

Date of amendment request: November 20, 1997.

Description of amendment request: The proposed amendment would revise the existing requirements for the Auxiliary Electrical Power Systems as identified in Technical Specifications (TSs) 3/4.10.A and TS 3.10.A.2.b. The specific changes are:

(1) The requirements in TS 3.10.A.2.b. are revised to omit the allowance for Spare Charger AB to substitute for either Charger A or B.

(2) The Bases in TS 3.10.A. are revised to omit the statements that justify Spare Charger AB to substitute for either Charger A or Charger B.

The proposed changes provide more limiting requirements for operation with the standby battery charger in service.

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

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Neither batteries, nor their chargers, are considered to be an initiator of any previously analyzed accident. Therefore, this change will not significantly increase the probability of any previously analyzed accident.

At least one Battery System is required to be available to mitigate the consequences of a Design Basis Accident. This change removes an allowance which places the unit in a more vulnerable condition through the unrestricted use of the spare battery charger. Since this change limits such a condition, it maintains the assumptions of the safety analysis, and therefore, will not significantly increase the consequences of any previously analyzed accident.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) nor is operation of the currently installed equipment changed. The change will, however, limit a currently allowed configuration with the spare charger and is more conservative. Thus, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

The proposed change continues to provide the previous margin of safety regarding the capability to withstand a single failure. At least one Battery System will continue to be available to provide the required safety function. The change will limit a currently allowed configuration with the spare charger and is thus more conservative. Therefore, this change will not significantly reduce a margin of safety.

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

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

NRC Project Director: Ronald Eaton, Acting Director.

Vermont Electric and Power Company

[Docket Nos. 50–280 and 50–281]

Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of amendment request: November 5, 1997.

Description of amendment request: The proposed change to Technical Specifications 5.3 and 5.4 would reflect an increase in the maximum permitted fuel enrichment to 4.3 weight percent U^{235} from the current 4.1 weight percent U^{235}. Fuel burnup limits and reactor operating power level would remain unchanged.

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:

Virginia Electric and Power Company has reviewed the Technical Specifications changes for Surry Units 1 and 2 against the criteria of 10 CFR 50.92. It has been concluded that use of fuel with the slightly higher initial enrichment does not involve a significant hazards consideration as defined in 10 CFR 50.92. An increase in the maximum initial fuel enrichment from 4.1 to 4.3 weight percent U^{235} will not:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The only accidents for which the probability of occurrence is potentially affected by the fuel enrichment involve criticality events during handling and storage. Analyses have demonstrated that the K-effective will be low enough to ensure subcriticality during both normal operation and under postulated accident conditions during the handling and storage of both new and spent fuel. Therefore, the probability of occurrence of criticality during fuel handling or storage is not increased. Safety analyses of record are based on inputs which bound the proposed increase in fuel enrichment. Since no changes to the fuel burnup limits are requested, the radiological consequences of previously evaluated accident scenarios will not be increased. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. Create the possibility for a new or different type of accident from any accident previously evaluated. Fuel with the higher initial enrichment will meet all applicable design criteria and will operate within existing Technical Specifications limits. Adherence to these standards and criteria precludes new challenges to components and systems that could introduce a new type of accident. All design and performance criteria will continue to be met. In addition, the use of a slightly higher initial fuel enrichment does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in the margin of safety. Surry Units 1 and 2 will continue to operate in compliance with the Technical Specifications, ensuring that the plants continue to provide acceptable levels of protection for the health and safety of the public. The Technical Specifications are based upon assumption[s] made in the safety and accident analyses, including those relating to the fuel enrichment and the design of the fuel storage areas. Analyses have demonstrated that subcriticality will be ensured during fuel storage and handling accident scenarios for both new and spent fuel. Additionally, safety analyses of record for core operation will remain applicable for Surry Unit 1 and 2 cores which use fuel with the slightly higher U^{235} enrichment. Therefore, the regulated margin of safety as defined in the Bases to the Surry Technical Specifications is not reduced.

Based on the preceding information, it has been determined that the use of fuel with an initial enrichment of up to 4.3 weight percent U^{235} satisfies the no significant hazards consideration criteria of 10 CFR 50.92.

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: Swern Library, College of William and Mary, Williamsburg, Virginia 23185.

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

NRC Project Director: James E. Lyons.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

Commonwealth Edison Company

[Docket Nos. STN 50–454 and STN 50–455]

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

Date of application for amendments: June 30, 1997, as supplemented on September 25, 1997.

Brief description of amendments: The amendments grant partial credit for boron in the spent fuel pools to maintain the subcriticality.

Date of issuance: December 4, 1997.

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

Amendment Nos.: 94, 94, 86 and 86.

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

Date of initial notice in **Federal Register***:* October 22, 1997 (62 FR 54868).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 4, 1997.

No significant hazards consideration comments received: No.

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

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 14, 1997, as supplemented. July 29, 1997, and August 13, 1997. The July 29, 1997, and August 13, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the May 7, 1997, **Federal Register** notice.

Brief description of amendments: These amendments relocate certain administrative control Technical Specifications (TSs) from the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS–1 and BVPS–2), TSs to the licensee's operational quality assurance program description, which is presented in Section 17.2 of the BVPS–2 Updated Final Safety Analysis Report (UFSAR). Section 17.2 of the BVPS–2 UFSAR contains the quality assurance program description for both BVPS–1 and BVPS–2. The following TSs are being relocated to the quality assurance program description.

BVPS–2 TS 6.2.3 (Independent Safety Evaluation Group)
BVPS–1 and BVPS–2 TS 6.5.1 (Onsite Safety Committee)
BVPS–1 and BVPS–2 TS 6.5.2 (Offsite Review Committee)
BVPS–1 and BVPS–2 TS 6.8.2 (Procedures, Review and)
BVPS–1 and BVPS–2 TS 6.8.3 (Temporary Procedure Changes, Review and Approval)
BVPS–1 and BVPS–2 TS 6.10.1 (Records Retention, At least 5 Years)
BVPS–1 and BVPS–2 TS 6.10.2 (Records Retention, Duration of Operating License)

Date of issuance: December 10, 1997.

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

Amendment Nos.: 209 and 87.

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

Date of initial notice in **Federal Register***:* May 7, 1997 (62 FR 24986).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 10, 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: July 17, 1996, as supplemented by letters dated June 3, and July 7, 1997. Also, application dated April 11, 1997.

Brief description of amendment: The amendment changes the Appendix A Technical Specification (TS) 3.7.1.3 by increasing the minimum required contained water volume in Condensate Storage Pool from 82 percent to 91 percent indicated level. In addition, this amendment expands the applicability of TS 3.7.1.3 to include Mode 4 operational requirements. The amendment also deletes Action (b) in TS 3.7.1.3 and its associated surveillance requirement in Waterford 3 TSs.

Date of issuance: December 18, 1997.

Effective date: December 18, 1997, to be implemented within 60 days.

Amendment No.: 137.

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

Date of initial notice in **Federal Register***:* March 26, 1997 (62 FR 14461), July 30, 1997 (62 FR 40849) and April 22, 1997 (62 FR 19624).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 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.

Florida Power Corporation, et al.

[Docket No. 50–302]

Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: August 26, 1997.

Brief description of amendment: The amendment involves a revision to the design basis of the Emergency Diesel Generator (EDG) Air Handling System at Crystal River 3 resulting from the EDG upgrade modification which increased the 200-hour and 2000-hour service ratings for each EDG.

Date of issuance: December 12, 1997.

Effective date: December 12, 1997.

Amendment No.: 160.

Facility Operating License No. DPR–31: Amendment revises the Final Safety Analysis Report.

Date of initial notice in **Federal Register**: September 24, 1997 (62 FR 50004).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal River, Florida 34428

Indiana Michigan Power Company

[Docket Nos. 50–315 and 50–316]

Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: September 19, 1997 (AEP:NRC:1278).

Brief description of amendments: The amendments modify Technical Specification 4.5.2.d.1 to delete the interlock that would close the Residual Heat Removal (RHR) suction valves if the Reactor Coolant System (RCS) pressure were to increase to 600 psig while retaining the interlock that would prevent the suction valves from opening while the RCS pressure is above the RHR system design pressure. This change maintains the open interlock function and allows continued deactivation of the isolation valves to assure RHR availability and provide low temperature overpressure protection.

Date of issuance: December 10, 1997.

Effective date: December 10, 1997, with full implementation within 45 days.

Amendment Nos.: 219 and 203.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 22, 1997 (62 FR 54861).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 10, 1997.

No significant hazards consideration comments received: No.

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

Pennsylvania Power and Light Company

[Docket Nos. 50–387 and 50–388]

Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: October 7, 1996, as supplemented by letter dated May 9, 1997.

Brief description of amendments: These amendments modify Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications Table 3.3.2–2 by revising the trip setpoints and allowable values for secondary containment isolation radiation monitors.

Date of issuance: December 8, 1997.

Effective date: December 8, 1997.

Amendment Nos.: 170 and 143.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 18, 1996 (61 FR 66716).

The May 9, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

Pennsylvania Power and Light Company

[Docket Nos. 50–387 and 50–388]

Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: April 4, 1997, as supplemented April 14, June 6, and September 2, 1997.

Brief description of amendments: These amendments clarify the scope of the surveillance requirements for response time testing of instrumentation in the reactor protection system, isolation actuation system, and emergency core cooling system in the Technical Specifications for each unit (Sections 4.3.1.3, 4.3.2.3, and 4.3.3.3).

Date of issuance: December 8, 1997.

Effective date: December 8, 1997.

Amendment Nos.: 171 and 144.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: April 17, 1997 (62 FR 17885).

The April 14, June 6, and September 2, 1997, letters provided clarifying information that did not change the original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, PA 18701.

Power Authority of the State of New York

[Docket No. 50–333]

James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: December 14, 1995, as supplemented September 26, 1997.

Brief description of amendment: The amendment proposes to change the James A. FitzPatrick Technical Specifications to incorporate the inservice testing requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

Date of issuance: December 2, 1997.

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

Amendment No.: 241.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 22, 1996 (61 FR 1635).

The September 26, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 2, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Rochester Gas and Electric Corporation

[Docket No. 50–244]

R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: September 29, 1997, as supplemented October 8, 1997.

Brief description of amendment: The amendment revises the Ginna Station Technical Specifications (TS) to allow referencing of revision of the Ginna Station pressure and temperature limits report for the reactor coolant system pressure and temperature limits and low temperature overpressure protection limits. The amendment also corrects a typographical error in the TSs.

Date of issuance: December 9, 1997.

Effective date: December 9, 1997.

Amendment No.: 70.

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

Date of initial notice in **Federal Register:** November 5, 1997 (62 FR 59921).

The September 29 and October 8, 1997, superseded in their entirety the applications dated December 13, 1996, April 24, 1997, and June 3, 1997.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 9, 1997.

No significant hazards consideration comments received: No.

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

Southern California Edison Company, et al.

[Docket Nos. 50–361 and 50–362]

San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: December 22, 1995, as supplemented by letter dated November 25, 1997.

Brief description of amendments: These amendments revise License Conditions 2.E and 2.G for the San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The amendments delete the physical protection program reporting requirement from License Condition 2.G, and clarify in License Condition 2.E that not all documents composing the physical protection program plans necessarily contain safeguards information.

Date of issuance: December 16, 1997.

Effective date: December 16, 1997.

Amendment Nos.: Unit 2—138; Unit 3—130.

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

Date of initial notice in **Federal Register:** November 5, 1997 (62 FR 59921). The November 25, 1997, letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 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.

Virginia Electric and Power Company, et al.

[Docket Nos. 50–338 and 50–339]

North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: May 14, 1997, as supplemented October 15, 1997. The October 15, 1997, submittal provided clarifying information only, and did not change the proposed no significant hazards consideration determination.

Brief description of amendments: The proposed action consists of changes to the Technical Specifications (TS) revising Surveillance Requirement 4.7.1.7.2.a for both units to clarify the testing and inspection methodology of the turbine governor control valves. The proposed changes also provide clarification in the TS Bases Section 3/4 7.1.7 for the Turbine Valve Freedom Testing of the turbine governor control valves.

Date of issuance: December 4, 1997.

Effective date: December 4, 1997.

Amendment Nos.: 207 and 188.

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40860).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 4, 1997.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 24th day of December 1997.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 97–33968 Filed 12–30–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Membership on the Executive Resources Board

AGENCY: Nuclear Regulatory Commission.

ACTION: Appointment to the Executive Resources Board for the Senior Executive Service.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has announced the following appointments to the NRC Executive Resources Board. The Executive Resources Board is responsible for providing institutional continuity in executive personnel management by overseeing NRC's Senior Executive Service (SES) and Senior Level System (SLS) succession planning, merit staffing, and position management activities.

Appointees

L. Joseph Callan, Executive Director for Operations, Chair

Karen D. Cyr, General Counsel

Anthony J. Galante, Chief Information Officer

Jesse L. Funches, Chief Financial Officer

Hugh L. Thompson, Jr., Deputy Executive Director for Regulatory Programs

Ashok C. Thadani, Acting Deputy Executive Director for Regulatory Effectiveness

Patricia G. Norry, Deputy Executive Director for Management Services

Samuel J. Collins, Director, Office of Nuclear Reactor Regulation

Carl J. Paperiello, Director, Office of Nuclear Material Safety and Safeguards

Malcolm R. Knapp, Acting Director, Office of Nuclear Regulatory Research

Timothy T. Martin, Director, Office for the Analysis and Evaluation of Operational Data

Edward L. Halman, Director, Office of Administration

Paul E. Bird, Director, Office of Human Resources

Irene P. Little, Director, Office of Small Business and Civil Rights

John C. Hoyle, Secretary of the Commission

Hubert J. Miller, Regional Administrator, Region I

Luis A. Reyes, Regional Administrator, Region II

A. Bill Beach, Regional Administrator, Region III

Ellis W. Merschoff, Regional Administrator, Region IV

EFFECTIVE DATE: December 18, 1997.

FOR FURTHER INFORMATION CONTACT: Carolyn J. Swanson, Secretary, Executive Resources Board, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 (301) 415–7530.

Dated at Rockville, Maryland, this 19th day of December, 1997.

For the U.S. Nuclear Regulatory Commission.

Paul E. Bird,

Director, Office of Human Resources.

[FR Doc. 97–34075 Filed 12–30–97; 8:45 am]

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