the licensee's presentation, followed in turn by an opportunity for the licensee to respond to the complainant's presentation. In cases where the complainant is unable to attend in person, arrangements will be made for the complainant's participation by telephone or an opportunity given for the complainant to submit a written response to the licensee's presentation. If the licensee chooses to forego an enforcement conference and, instead, responds to the NRC's findings in writing, the complainant will be provided the opportunity to submit written comments on the licensee's response. For cases involving potential discrimination by a contractor or vendor to the licensee, any associated predecisional enforcement conference with the contractor or vendor would be handled similarly. These arrangements for complainant participation in the predecisional enforcement conference are not to be conducted or viewed in any respect as an adjudicatory hearing. The purpose of the complainant's participation is to provide information to the NRC to assist it in its enforcement deliberations.

* * * * *

Dated at Rockville, Maryland, this 3rd day of October 1997.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 97–26690 Filed 10–7–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Nuclear Waste; Notice of Meeting

The Advisory Committee on Nuclear Waste (ACNW) will hold its 95th meeting on October 21, 1997, at the William F. Bolger Center For Leadership Development, 9600 Newbridge Drive, Potomac, Maryland, and October 22–23, 1997, in Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance. The schedule for this meeting is as follows:

Tuesday, October 21, 1997–8:30 a.m. until 6:00 p.m.

Wednesday, October 22, 1997–8:30 a.m. until 6:00 p.m.

Thursday, October 23, 1997–8:30 a.m. until 4:00 p.m.

A. *ACNW Retreat*—The Committee members will discuss their mission, planned accomplishments, priorities, and work processes for FY 1998–99. The retreat will be held on October 21, 1997, at the William F. Bolger Center For Leadership Development.

B. *Meeting with NRC's Director, Division of Waste Management, Office of Nuclear Material Safety and Safeguards*—The Committee will meet with the Director to discuss technical assistance, developments at the Yucca Mountain project, resources, and other items of mutual interest.

C. *Review of NRC Research and Technical Assistance*—The Committee will review activities of NRC's Office of Nuclear Materials Safety and Safeguards and Nuclear Regulatory Research in the area of nuclear waste disposal. The ACNW will provide input to the Advisory Committee on Reactor Safeguards' February 1998 report to Congress on NRC research.

D. *Prepare for Next Meeting with the Commission*—The Committee will prepare for its next formal meeting with the Commission. The Committee is scheduled to discuss items of mutual interest with the Commission on December 17, 1997.

E. *Preparation of ACNW Reports*—The Committee will discuss planned reports, including a recommended approach to implement the defense-in-depth concept in the revised 10 CFR Part 60, the Application of Probabilistic Risk Assessment Methods to Performance Assessment in the NRC High-Level Waste Program, ACNW priority issues for 1998, and other topics discussed during the meeting as the need arises.

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

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

Procedures for the conduct of and participation in ACNW meetings were published in the **Federal Register** on September 2, 1997 (62 FR 46382). In accordance with these procedures, oral or written statements may be presented by members of the public, electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify the Chief, Nuclear Waste Branch, Mr. Richard K. Major, as far in advance as practicable so that appropriate arrangements can be made to schedule the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Chief, Nuclear Waste Branch, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Major as to their particular needs.

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

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

Dated: October 2, 1997.

John C. Hoyle,

Acting Advisory Committee Management Officer.

[FR Doc. 97–26692 Filed 10–7–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the

Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 15, 1997, through September 26, 1997. The last biweekly notice was published on September 24, 1997 (62 FR 50000).

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 November 7, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: 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.

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

Date of amendments request: March 18, 1997, as supplemented by letters dated July 28, 1997 and September 9, 1997

Description of amendments request: The amendments would revise the operating licenses for Palo Verde Units 1, 2 and 3 to reflect approval of Amendment 42 to the Palo Verde Nuclear Generating Station (PVNGS) Physical Security Plan. Amendment 42 would revise the methods used to search materials, packages and personnel prior to their entry into the protected area, as described within the security plan.

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

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

The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated. The ''accident'' as it relates to the Security Plan would have to be an impact to the Design Basis Threat (DBT) postulated for PVNGS. This change does not decrease the overall security systems (as described in paragraph's (b) through (h) of 10 CFR 73.55) ability to protect PVNGS with the objective of high assurance against the DBT of radiological sabotage as stated in 73.1(a). This change does not delete or contradict any regulatory requirements.

The applicable design basis threat is described in 10 CFR 73.1. Based on that threat, the probability of an external determined violent assault by stealth, or deceptive actions, of several persons is unaffected by the requested changes to the search requirements. Similarly, an internal threat of an insider, including an employee (in any position) is no more likely to occur as a result of the search techniques. The probability of an attack with a four-wheel drive land vehicle bomb is unaffected. Theft or diversion of formula quantities of strategic special nuclear material is a threat of removal from the inside of the protected area, which is not within the scope of this change that only affects searches of material entering the protected area.

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

The possibility of an accident of a new or different kind has not been created because the DBT (as described in the Security Plan and 10 CFR 73.1) would not be changed as a result of these changes. The changes supplement regulatory requirements and commitments already described in the PVNGS Physical Security Plan.

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

The proposed changes do not involve a significant reduction in a margin of safety. These changes to the personnel, material and package search criteria are not specifically considered in the basis for any margin of safety. The DBT considers inside assistance by a knowledgeable individual, however, these changes would not assist this individual in either sabotage or theft of nuclear material.

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

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

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

NRC Project Director: William H. Bateman

Duke Energy Corporation, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: September 15, 1997

Description of amendment request: The proposed license amendments would revise the Technical Specifications (TS) to:

1. Revise the reactor coolant system heatup limitation curves in Figure 3.4-2, which are applicable only to the first 10 effective full-power years (EFPYs). The revised curves would be (a) applicable to the first 15 EFPYs; (b) include the latest radiation surveillance capsule results; (c) remove instrument margins by relocating them to a licensee-controlled document, ''Pressure Temperature Limit Report;'' and (d) administratively delete certain unneeded footnotes that exist in the current figure.

2. Modify the actual surveillance capsule identification listed in Table 4.4-5, ''Reactor Vessel Material Surveillance Program - Withdrawal Schedule'' (for Unit 2 only) and update each unit—s lead factors and withdrawal time.

3. Revise the power-operated relief valve (PORV) setpoints in Section 3.4.9.3.a to less than or equal to 400 pounds per square inch gauge (psig) (as left calibrated), allowable value less than or equal to 425 psig (as found).

4. Make editorial changes to improve consistency among various TS sections to conform with the Westinghouse Improved Standard Technical Specifications, and update applicable Code references.

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

1. Will the changes involve a significant increase in the probability or consequences of an accident previously evaluated?No. No previously evaluated accident was considered to originate from use of the heatup curves (change 1. above), the testing and use of surveillance capsules (change 2. above), the setpoint of PORVs (change 3. above), and editorial changes to the TS. Also, these items did not have any role in previously analyzed accident scenarios and thus no impact on accident consequences. Therefore, these proposed changes will have

no impact on the consequences or probabilities of any type of previously evaluated accidents.

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

No. No actual plant equipment or operating procedure will be affected by the proposed changes. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

3. Will the changes involve a significant reduction in a margin of safety?

No. The margin of safety is associated with confidence in the design and operation of the plant. The changes to the TS do not involve any change to plant design or operation. Thus, the margin of safety previously analyzed and evaluated is maintained.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: June 14, 1997

Description of amendment request: The proposed amendment would revise the technical specifications (TS) for the Crystal River Nuclear Electric Generating Plant Unit 3 (CR-3). The proposed TS changes reflect the operational limitations in mitigating certain Small break loss-of-coolant-accident (SBLOCA) events. The licensee also proposed changes to the associated licensing and design bases.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below. The proposed changes are addressed in three major parts: (1) SBLOCA Mitigation, (2) Emergency Diesel generator (EDG) upgrade and (3) EDG Load Rejection Test and Steady State Loads.

SBLOCA Mitigation

The licensee's revised SBLOCA analyses show that for certain sized breaks, a combination of emergency core cooling system (ECCS) flow to the reactor vessel and emergency feedwater (EFW) flow to the once through steam generators (OTSG) is needed to provide for adequate core decay heat removal. Due to load capacity limits on the —A— EDG, the length of time that the motor-driven emergency feed pump-1 (EFP-1) would be available is limited. To ensure adequate EFW system flow and core decay heat removal, several actions would have to be initiated. They include —A— EDG load management, and EFW flow through the turbine-driven emergency feedwater pump-2 (EFP-2) by opening the cross tie valve, flow through both the high pressure injection (HPI) pumps and EFP-1. The proposed TS changes reflect the operational limitations and other associated required actions to ensure adequate ECCS and EFW cooling capability remains. These changes for system cross train dependencies and EDG load management are required for the remainder of current Cycle 11 only.

1. The proposed Technical Specification changes, modifications, and operator actions involving SBLOCA mitigation will not result in a significant increase in the probability of an accident previously evaluated. In addition, the portions of the change involving cross-train dependencies and load management are being requested for the remainder of Cycle 11 only, which limits the impact on any previously established probabilities. The initiators of any design basis accident is not affected by the proposed Technical Specification changes, modifications, and operator actions involving SBLOCA mitigation. Consequently, there is no significant impact on any previously evaluated accident probabilities.

The proposed Technical Specification changes, modifications and operator actions involving SBLOCA mitigation do not result in a significant increase in the consequences of SBLOCA mitigation-related accidents previously evaluated. In this regard, the proposed Technical Specification changes, modifications and operator actions will not adversely affect the integrated ability of the EDGs and the EFW, SW [service water], RW [raw water], Control Complex Cooling, ECCS, DC [Decay Heat Closed Cycle Cooling Water System], Decay Heat Seawater, and Electrical Distribution Systems to perform their intended safety functions. Therefore, the combined ability of these components and systems and actions to mitigate the consequences of a SBLOCA will continue to be maintained. In fact, the collective impact of these Technical Specification changes, modifications and operator actions represents a restoration of the ability to mitigate the consequences of a SBLOCA, which are consistent with the consequences assumed in licensing and design basis for CR-3. For example, the installation of EFW cavitating venturis and the improved operational range of the turbine driven feedwater pump increase the ability of the EFW system to mitigate the consequences of a SBLOCA. In addition, the Technical Specification changes, modifications and operator actions do not significantly affect the onsite or offsite doses which remain a small fraction of 10 CFR Part 100 limits.

2. The proposed Technical Specification changes, modifications and operator actions do not create the possibility of a new or different kind of accident from any accident previously evaluated. The Technical Specification changes, modifications, and operator actions do not involve a different initiator for any design basis accident and do not create new design basis scenarios. SBLOCA mitigation, utilizing a combination of automatic and manual actions, is already part of the CR-3 licensing basis. Manual operator actions necessary for the mitigation of SBLOCAs are currently addressed or are being addressed in EOPs [emergency operating procedures]. Also, these Technical Specification changes, modifications and operator actions restore the ability to mitigate the impact of a SBLOCA, which is consistent with the CR-3 licensing and design basis. Based on the above, a new or different kind of accident does not result from this submittal.

3. The proposed Technical Specification changes, modifications and operator actions do not involve a significant reduction in the margin of safety for SBLOCA mitigation. The Technical Specification changes, modifications and operator actions for the EDGs and the EFW, SW, RW, Control Complex Cooling Systems represent a restoration of the overall margin of safety to a degree that it will be consistent with the existing plant design and licensing bases for SBLOCA mitigation.

EDG upgrade

This aspect of the proposed license amendment involves increases in the service ratings of the EDGs. The required amount of fuel oil in the EDG fuel day tank and fuel storage tank, and lube oil storage is being increased to ensure that adequate volume is available to support the new service ratings. The EDG refueling interval load test parameters are being revised to reflect the increased service ratings and to ensure that the minimum test load is equal to or greater than the expected maximum steady state accident load. Additionally, associated EDG Surveillance Requirements (SR) Bases are being revised.

1. The proposed Technical Specification changes, modifications and operator actions do not involve a significant increase in the probability of an accident previously evaluated because neither the EDGs nor the EDG—s fuel oil and lube oil systems serve as the initiator for any design basis accident and, therefore, do not significantly impact any previously evaluated accident probabilities.

The proposed Technical Specification changes, modifications and operator actions do not involve a significant increase in the consequences of an accident previously evaluated because the ability of the EDGs and the EDG fuel oil and lube oil to perform their intended safety function has not been adversely affected. The EDGs and the EDG fuel oil and lube oil systems remain fully capable of performing their safety function for all design basis accidents. The increase in loading permitted under these changes will reflect the manufacturer—s certified capabilities of the EDGs. Also, the increase in the required fuel remains within the capabilities of the fuel tanks. The same potential design basis failures that existed prior to the EDG upgrades will continue to exist subsequent to the modifications. It follows that the consequences of such failures will remain a small fraction of 10 CFR Part 100 limits.

2. The proposed Technical Specification changes, modifications and operator actions do not create the possibility of a new or different kind of accident from any accident previously evaluated. Also, the proposed Technical Specification changes, modifications and operator actions do not involve any new accident initiators, or a new or different kind of accident from any previously evaluated. In addition, the configuration and basic function of the EDGs and EDG's fuel and lube oil systems are unaffected by the changes. In fact, the EDG upgrades ensure that the previously evaluated accidents are consistent with system and component capabilities and the current design and licensing bases.

3. The proposed Technical Specification changes, modifications and operator actions do not involve a significant reduction in the margin of safety. The EDGs and EDG—s fuel and lube oil systems will continue to able to be perform their safety function for all design basis accidents. There is an increase in the net margin of safety for fuel and lube oil storage since required volumes have been recalculated and increased, additional margin has been added to the calculated results, and the required volumes are based on usable tank volumes instead of tank capacity. These volumes continue to bound the postulated worse-case accident scenario. The increase in fuel storage required by the changes remains within the capacity of the storage tanks. The Technical Specification changes, modifications and operator actions further ensure that margins provided in current design and licensing bases are satisfied.

EDG Load Rejection Test and Steady State Loads

The proposed changes for this part affects the TS Bases. The basis of the EDG load rejection test is being revised to bound the largest single load. A description of ''steady state'' is being provided with examples of short duration loads and loads imposed by the starting of motors. Also, addressed is the licensee's conclusion that the refueling interval EDG load test is not invalidated by loads imposed by the starting of motors.

1. The proposed Technical Specification changes, modifications and operator actions do not involve a significant increase in the probability of an accident previously evaluated because the EDG load tests and load rejection test do not serve as the initiator for any design basis accident and, therefore, do not significantly impact any previously evaluated probabilities.

The proposed Technical Specification changes, modifications and operator actions do not involve a significant increase in the consequences of an accident previously evaluated because the changes do not affect the ability of the EDGs to perform their intended safety function. Rather, the Technical Specification changes, modifications and operator actions provide further assurance that the EDGs are capable of performing their safety function. Failure of an EDG has the same consequences as it would if the changes were not made. It follows that the 10 CFR Part 100 consequences of such failures has not changed.

2. The proposed Technical Specification changes, modifications and operator actions do not create the possibility of a new or different kind of accident from any accident previously evaluated because the changes do not affect the ability of the EDGs to perform their intended safety function. The configuration and basic function of the EDGs, including accurately describing the manufacturer certified EDGs service ratings and steady state loads, do no create a possibility for a new or different kind of accident. Although the load rejection test is for an increased EDG largest single load, the kind of accident addressed by both the load rejection test and the refueling load test remain the same.

3. The proposed Technical Specification changes, modifications and operator actions do not involve a significant reduction in the margin of safety. The calculated loads imposed by the starting of motors are short duration, have a low probability of occurrence, and are expected to be within the manufacturer limits. In fact, the margin confirmed by EDG refueling load testing and load rejection testing will demonstrate a restoration of design and licensing margin and confirm that the EDGs remain fully capable of performing their safety function for all design basis accidents.

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: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC - A5A, P. O. Box 14042, St. Petersburg, Florida 33733-4042

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: September 16, 1997

Description of amendment request: The proposed changes to the Technical Specifications (TSs) would modify TS 3.7.1.1, ''Plant Systems Turbine Cycle Safety Valves.'' During its effort to verify the current design and licensing bases for Millstone, Unit 2, NNECO has determined that the maximum allowable power level high trip setpoints with inoperable steam line code safety valves specified in Table 3.7-1 of TS 3.7.1.1 are incorrect. The trip setpoints were not changed to be consistent with a previously approved reduction in the maximum power level high trip setpoint. In addition, NNECO is also in the process of reanalyzing the inadvertent closure of the main steam isolation valve (MSIV) and the loss of electrical load events. The results of the reanalysis indicate that the MSIV event results in the highest peak pressure in the secondary system and that the formula currently contained in the TS Bases for TS 3.7.1.1 may not result in the correct trip setpoints.

Specifically, NNECO proposes to: (1) delete TS Table 3.7.1 by not allowing operation in Mode 1 or 2 with inoperable steam line code safety valves, (2) modify the associated action statement in TS 3.7.1.1, and (3) update the TS Bases to reflect the proposed changes and update the amendment history numbers to reflect previously approved amendments.

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

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

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

This proposed change will remove the ability to operate in Modes 1 or 2 with inoperable main steam line code safety valves. Operation in Mode 3 will be retained, provided no more than three main steam line code safety valves per steam generator are inoperable.

The primary function of the main steam line code safety valves is to prevent secondary system overpressurization. These valves will also provide reactor core heat removal and design basis accident mitigation. This proposed change does not affect the length of time the plant can operate with inoperable main steam line code safety valves before compensatory actions must be taken. (Four hours is still allowed to restore the valve(s) to operable status.) This proposed change does not affect the probability of occurrence of any design basis accident and does not affect how the main steam line code safety valves function to mitigate design basis accidents. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the way any structure, system, or component functions. The proposed change will conservatively change plant operation in Modes 1 and 2 by removing the ability to operate at power with inoperable main steam line code safety valves as currently specified in Technical Specification 3.7.1.1. It does not introduce any new failure modes and does not alter any assumption made in the safety analysis.

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

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

This proposed change to Technical Specification 3.7.1.1 will remove the ability to operate in Modes 1 or 2 with inoperable main steam line code safety valves. Operation in Mode 3 will be retained, provided no more than three main steam line code safety valves per steam generator are inoperable. The operability of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psig) of the design pressure of 1000 psig during the most severe anticipated system operational transient. This change will not affect the operability requirements for the main steam line code safety valves and will not affect the length of time the plant can operate with inoperable main steam line code safety valves before compensatory actions must be taken. This will ensure the plant equipment required for design basis accident mitigation will be available. Therefore, there is no significant reduction in a margin of safety as defined in the Bases of Technical Specification 3.7.1.1.

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: August 29, 1997

Description of amendment request: Based on a review and subsequent calculations of the cold overpressurization protection (COPS) enabling temperature and the emergency core cooling system (ECCS)/ charging system Mode 3 requirements, NNECO proposes to reduce the COPS enabling temperature. As a result, NNECO proposed the following Technical Specifications (TS) changes: new heatup and cooldown pressure/ temperature limit curves and their associated requirements; new power operated relief valve (PORV) setpoint curves and their associated requirements; revisions to the reactor coolant loops and coolant circulation, ECCS, boration systems, and COPS to incorporate the lower enabling temperature and new restrictions for cold overpressure protection system (COPPS), PORV undershoot, and residual heat removal (RHR) relief valve bellows; addition of a footnote to allow a reactor coolant pump (RCP) to substitute for an RHR pump during heatup from Mode 5 to Mode 4, which is consistent with the improved standard technical specification (STS); reword TS 3/4.4.9.3 and its Bases section to be consistent with the improved STS; and revision of the affected Bases sections to be consistent with the proposed changes.

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

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

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

Probability of Occurrence of Previously Evaluated Accidents

Since the PORV setpoints and the COPS enabling temperature have been calculated in accordance with 10CFR50, Appendix G and ASME [American Society of Mechanical Engineers] Section XI, the change will not alter the probability that an overpressurization event will result in a loss of RV [reactor vessel] integrity. The new PORV setpoint curves are lower than the current curves in certain temperature ranges (below approximately 130—F and above approximately 220°F), and therefore the operating window is slightly decreased. However, the reduced operating window is still sufficient for normal anticipated pressure fluctuations. Below 160°F, operation of Reactor Coolant Pumps are prohibited if the PORVs are armed for COPPS; therefore, PORV actuation will not occur below 160—F when the RCPs are running. In a water solid condition, RCS [reactor coolant system] pressure is maintained via the letdown low pressure control valve, which, when in automatic mode, maintains the RCS pressure in a relatively narrow range. When the RCPs are not running, the PORV COPPS system can be actuated. However, for this condition, the allowable pressure range is 0 to 418 psia [pounds per square inch atmospheric]. This pressure range is sufficient to accommodate normal anticipated pressure fluctuations.

Above 220°F, the minimum pressure range is from 300 psia to 595 psia; this range is sufficient to accommodate normal anticipated pressure fluctuations. In this temperature range, a pressurizer bubble is normally present, which will minimize any pressure fluctuations, thereby limiting the possibility of a PORV actuation. Based on this, it is concluded that the proposed change will not impact the probability of occurrence that a PORV will be challenged.

When the RHR relief valves are used for COPS there is no credible scenario which would result in excessive relief valve undershoot. This is because these valves are spring loaded relief valves which are designed to close whenever the RCS pressure decreases below the nominal setpoint of 440 psig [pounds per square inch gauge]. This provides assurance that there will be no damage to the seal of a running RCP.

The proposed changes to the heatup/ cooldown curves and the reduction in the enabling temperature for COPS only affect operational limits and can not be initiators of an event. The restrictions on RC [reactor coolant], RHR and ECCS pump operation can not result in an event initiator. Two separate operator actions are required to start an ECCS or RC pump. These two necessary actions as well as procedural controls are sufficient to prevent an inadvertent ECCS or RC pump start. De-energizing the RCPs when returning a loop to service can not initiate an event.

The proposed change will provide an operable charging pump to ensure RCP seal flow and reactivity control will be available. When the RCP is in operation, the charging pump provides the preferred method for seal flow. The proposed change minimizes the time that this preferred method is interrupted. A loss of charging pump seal flow will not cause a malfunction of an RCP because the pump is designed to use RCS flow as an alternate method at these conditions. Not allowing two charging pumps to run simultaneously and requiring at least one pump to be in pull-to-lock, assures a second pump will not start on an inadvertent SI [safety injection] and exceed the assumptions in the Appendix G analysis or initiate a Boron Dilution or CVCS [chemical and volume control system] Malfunction event. If an operator were to inadvertently start the second pump, a failure of the charging throttle valve, FCV-121, and one relief valve credited for COPS would be necessary to exceed the assumptions in the Appendix G analysis. In addition, the actual time allowed for swapping the charging pumps is short. The remainder of the hour allows for documented verification of the disabling of the required pump. The proposed change will not change any control systems for these pumps or alter the system configuration that would affect the probability of an uncontrolled increase in charging flow. The procedure requirements to swap pumps and the likelihood of these multiple failures occurring during the short duration allowed in this footnote provide adequate assurance that an overpressurization event will not occur. Maintaining at least one pump always operable makes the system more reliable for reactivity control than the current method which disables both pumps simultaneously.

The proposed change to maintain one charging pump operable in Mode 4 [cannot] initiate an event because of the stable reactivity condition of the reactor, the emergency power supply requirement for the operable charging pump, and the fact that the plant is procedurally required to be borated to the highest required boron concentration for Modes 3, 4, or 5 prior to entering Mode 4. These changes do not effectively change the availability of plant equipment or the way that the plant is operated.

The proposed change to substitute an RCS loop for an RHR loop during a planned heatup, can not initiate an event. The RCP will be verified as operating properly prior to stopping the RHR pump and as such will not initiate a loss of decay heat removal (by heating up to steam the SGs [steam generators])/loss of flow. While the RCP is in operation, it performs the RHR boron mixing function and the decay heat removal function is not required for heatup. Using the RCP to perform this function will not affect the probability that the RCP could fail because it will be operated within its normal operating design conditions. Aligning RHR in the ECCS lineup will not affect the probability of a RHR pump to start. The pump will be operable in this lineup. Currently in Mode 5, RHR is lost on a LOP [loss of offsite power] and is manually restarted once the diesel is running. With the proposed change, the RCP will be lost on a LOP and the RHR pump will have to be manually started. Thus, the proposed change does not affect the probability that the RHR pump could fail. Since the current response to a LOP is to manually restart the RHR pump, operator action is needed independent of this change. The proposed change allows normally open valves to be closed in Mode 5 to align RHR for ECCS injection. This introduces additional manual actions which could extend the time required to establish flow. In addition, if one diesel generator were to fail, manual operation of a valve in the ESF [engineered safety features] building would be necessary. The mechanistic 'failure to open' of valves that is introduced by the change as well as the need for manual operator action to realign these valves increases the time to establish heat removal. However, there is sufficient time to re-establish RHR because this note applies only for a heatup in which the plant will have been shutdown for at least several hours which causes decay heat to be low (as compared to high decay heat immediately following a plant trip). Thus, it is concluded that there is no impact on the probability of failure of RHR to perform its required function.

The proposed change to the ECCS wording does not result in any new failure modes that could initiate an event since manual realignment from the control room is currently allowed. Nor can the manual alignment of RHR valves initiate an event because this alignment is only for accident mitigation.

Therefore, the proposed changes do not increase the probability of occurrence of previously evaluated accidents.

Consequences of Previously Evaluated Accidents

The revised Pressure/Temperature curves were calculated in accordance with 10CFR50, Appendix G, ASME Section XI, and Regulatory Guide 1.99, Revision 2. This provides assurance that an inadvertent overpressurization event will not result in a loss of RV integrity. The restrictions on RCP operation and the requirement to de-energize the RCPs in Modes 5 and 6 when returning a loop to service are consistent with the assumptions made in this Appendix G analysis and the RCPs are not required for accident mitigation for any previously evaluated accidents and therefore do not affect the consequences.

The COPS relieving capability is greater than the maximum RCS pressurization rate resulting from any allowed pump combinations, and the PORV setpoints have been adjusted to take into account instrumentation effects. This will provide assurance that COPS will continue to perform its safety function. Since the COPS enabling temperature has been demonstrated to be conservative at 275—F, allowing SI pump operability above 275—F will have no impact on vessel non-ductile failure.

The restriction between 275—F and 350—F on the SI and charging pumps, has been appropriately moved to the reactor coolant loop section to provide protection for the RHR system (RCS protective boundary) and to the cold overpressure protection section to provide protection for the RHR relief valves and the RCP seals. By incorporating this requirement previously located in the ECCS TS, RCS integrity is ensured.

With the RCS less than 160—F, the consequences of the PORV undershoot from the proposed PORV setpoints are that the RCS pressures may drop below the minimum requirement for RCP seal integrity. However, no seal damage will occur since a requirement has been added prohibiting the operation of RCPs below 160°F with the PORVs not isolated while in the low setpoint mode. With cold overpressure relief valves in service above the COPS enable temperature (275°F), restrictions are placed on the startup of an RCP and the number of ECCS pumps capable of injecting into the RCS to prevent unacceptable mass or energy addition transients. This provides assurance that the RHR relief valve capacity will not be exceeded and that PORV undershoot will not challenge the RCP ι1 seal. The restriction on the maximum number of ECCS pumps ensures that the integrity of the RHR relief valve bellows and the RCP seals during mass injection transients (i.e., inadvertent SI).

The restrictions on RCS/SG secondary side temperature mismatch ensure that an unanalyzed energy addition event does not occur when an RCS loop is placed in operation.

The consequences of a small break LOCA [loss of coolant accident] in COPS Mode 4 are not affected because the plant will continue to maintain one charging pump operable in Mode 4. In addition, additional options are provided in the bases of TS 3/4.4.9.3 for disabling the required charging and SI pumps that will allow faster restoration if required to mitigate a LOCA or loss of RHR in Modes 4, 5 and 6.

An RHR pump will remain available in Mode 4 with manual realignment from the control room as required to perform its ECCS safety function. The changes have no impact on the capability of RHR to function in the ECCS mode. RHR is credited during a safety grade cold shutdown. The proposed change assures that the RHR system will be available to perform its heat removal function during a safety grade cold shutdown and thus, there is no change in the analysis assumptions or consequences.

The changes also eliminate an inconsistency between the charging system operability requirements for boration and the charging system operability requirements for cold overpressure protection. The requirement to maintain two charging pumps operable in Mode 4 will be reduced to one charging pump. As stated in the proposed basis section, a second method of boration is not required to be OPERABLE in Mode 4 for single failure considerations based on the stable reactivity condition of the reactor, the emergency power supply requirement for the operable charging pump, and the fact that the plant is procedurally required to be borated to the highest required boron concentration for Modes 3, 4, or 5 prior to entering Mode 4. This provides assurance that reactivity control will be maintained and stable while only one charging pump is operable for cold overpressure concerns. These changes do not effectively change the availability of plant equipment or the way that the plant is operated. The changes will not adversely impact the assumption for the limiting dilution flow path and flow rate and therefore, the consequences of a boron dilution event are not affected.

The proposed changes will maintain a charging pump operable for reactivity control while ensuring that the flow limits in the Appendix G analyses are not exceeded. Remaining within the bounds of the Appendix G limits ensures reactor vessel integrity in Mode 4. Since the change maintains the reactor vessel integrity, it does not introduce any means of releasing radionuclides post-accident. The consequences of a small break LOCA in Mode 4 are not affected because the plant will continue to maintain one charging pump operable in Mode 4. These changes are reflected in TS 3.1.2.1, 3.1.2.2, 3.1.2.3 and 3.1.2.4. Adequate protection is provided for reactor vessel integrity while maintaining reactivity control operability.

In Mode 5, RHR requirements are specified for decay heat removal in the case of a loss of offsite power but none are specified for ECCS accident mitigation. The first RHR train will be aligned for injection prior to taking the second train out of service. This provides assurance that this train will be available if needed in Mode 5. Currently in Mode 5, following a LOP the RHR system can be re-established by restarting the RHR pump once the diesel is running. No valve manipulations are necessary. With the proposed change, when the operating RCP trips following a LOP, some of the RHR valves must be realigned from the ECCS to heat removal mode. If one diesel generator were to fail, manual operation of a valve in the ESF building would be necessary. Since this footnote is only applicable during a heatup, decay heat will be low. There is sufficient time to re-establish RHR even if action outside the control room is necessary. Since there are four operable RCS loops, a bubble drawn in the pressurizer and the RCS pressurized, the plant will heat up to Mode 4 and natural circulation will provide core cooling if the RHR system cannot be re-established. Thus, decay heat removal is assured and there is no affect on the consequences of a LOP.

Since the structural integrity of the RCS is maintained and adequate core cooling and reactivity control will be available for design basis events, the proposed changes will have no adverse impact on the consequences of previously evaluated accidents.

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

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

The temperature/pressure limits will continue to meet the requirements of 10CFR50, Appendix G. Since the new limits continue to provide assurance of reactor vessel integrity, the proposed change does not create the possibility of an accident of a different type than previously evaluated. Adequate RCS pressure-relieving capabilities will continue to be maintained throughout the shutdown modes. No new malfunctions will be introduced which could result in a new accident postulated in Modes 3-5.

The restrictions on RCP operation do not create the potential for unanalyzed heat injection transient as a result of an inadvertent RCP start because two operator actions are required to start a pump. The requirement to have all RCPs de-energized, prior to unisolating a loop adds additional assurance that an energy addition transient will not occur.

The proposed change to allow 2 charging pumps to be operable does not create an accident of a different type because there will be adequate controls to ensure that the second pump does not inadvertently start and initiate an increase in RCS inventory or a boron dilution. Procedural controls will minimize the amount of time that both charging pumps are operable and at no time will two pumps be out of pull-to-lock.

The proposed footnote to TS 3.4.1.4.1 to remove RHR heat removal from operation allows normally open valves to be closed in Mode 5 to align RHR for ECCS injection. This introduces 'failure to open' as a potential mechanistic failure malfunction in the RHR system. This is a malfunction of a different type since previously stroking of these valves was not needed to establish RHR. The current response to a LOP is to manually restart the RHR pump only, with no valve manipulations required. The proposed change adds the manual action of realigning the valves. Since operator action to re-establish RHR following a LOP is required independent of the proposed changes, crediting operator action does not create the potential for a malfunction of a different type. Allowing both trains of RHR to be out of service does not create a different accident because additional requirements have been specified for RCS loop operability and at least one RHR pump is operable for ECCS when the core cooling requirement is being met by crediting RCS loop operability. Meeting the Mode 4 TS conditions prior to heatup, ensures two diesels are operable. As such, a single failure would only require one valve to be manually realigned in the ESF building. Adequate time is available to accomplish these actions since this note only applies during heatup, when decay heat is very low. Further, with four RCS loops operable and a bubble drawn in the pressurizer and the RCS pressurized, the steam generators can be used for core cooling via natural circulation once the plant heats up to Mode 4, in the event the RHR cannot be re-established. Since core cooling will be assured if a LOP occurred during heatup in Mode 5, the change in plant response to this event does not constitute an accident of a different type.

The proposed changes to TS 3.5.3.f to manually realign the ECCS valves is no different from what is currently evaluated. During a Mode 4 LOCA adequate procedural guidance is provided to ensure that RHR will be realigned for injection. The proposed change allows RHR to be aligned to perform its safety grade cold shutdown heat removal function.

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

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

The new proposed curves raises the lower bound on RCS temperature, resulting in increased RCS ductility and therefore increased structural margin against non-ductile failure. The new curves take into account the dynamic pressure effects identified in NRC Information Notice 93-58 and are calculated in accordance with 10CFR50 Appendix G, ASME Section XI and Regulatory Guide 1.99, Revision 2. These changes to the P/T [pressure/temperature] limits are reflected in TS 3.4.9.1. Additional restrictions have been placed on RCP operation to ensure that assumptions used in developing the curves remain bounding. These are also reflected in TS 3.4.1.3, 3.4.1.4.1, 3.4.1.4.2 and 3.4.1.6. As such, the curves will continue to provide the required assurance for reactor vessel integrity.

The COPS enable temperature is proposed to be lowered from the current 350°F to 275°F which provides a margin of 31—F above that required by NRC Branch Technical Position RSB 5-2. The reduction of the COPS enabling temperature eliminates the need for COPS to be operable in Mode 3. This will simplify the transition between Mode 3 and Mode 4.

Additional changes have been made to the Overpressure Protection TS to ensure that the assumptions made in the Appendix G calculations remain bounding. These include additional restrictions on charging pump and SI pump operability and the modification of the PORV setpoints. The pump requirements have been transferred from the ECCS specification and expanded to cover Modes 4, 5 and 6. In addition, these same pump restrictions have been included in TS 3.4.1.3 whenever RHR is in service. This provides added assurance that the RHR piping will not be overpressurized by an inadvertent actuation of an SI or charging pump. Additional actions and surveillances have been provided to assure that assumptions on charging pump and SI pump operability will be met. The additional options for assuring the inoperability of the SI and charging pumps require two distinct operator actions to restore injection capability from these pumps. Thus, these options are equivalent in providing assurance that an inadvertent injection will not occur while at the same time allowing faster restoration if needed to mitigate a loss of RHR.

A requirement to have all RCPs de-energized, prior to unisolating a loop is added to TS 3.4.1.6.c, to ensure that loop flow will not be initiated which results in an energy addition transient from the secondary side of the SG being unisolated. This change will preclude RCS overpressurization when an idled loop is returned to service and SG secondary side temperature is greater than the RCS temperature.

The PORV setpoints were established to ensure that the P/T limit curves are not exceeded as a result of a single operator action or as a result of a single equipment malfunction, as required by the current system design basis criteria (i.e., SRP [standard review plan] Branch Technical Position RSB 5-2).

A clarification of the hydrostatic and leak test requirements ensures a uniform reactor vessel temperature for the test. A 72 hour time limit is placed on the performance of engineering evaluations of out of specification condition. This provides added assurance for RPV [reactor pressure vessel] integrity.

The changes also eliminate an inconsistency between the charging system operability requirements for boration and the charging system operability requirements for cold overpressure protection. These are reflected in TS 3.1.2.1, 3.1.2.2, 3.1.2.3 and 3.1.2.4. The Bases requirement to maintain two charging pumps operable in Mode 4 will be reduced to one charging pump. As stated in the proposed basis section, a second method of boration is not required to be OPERABLE in Mode 4 for single failure considerations based on the stable reactivity condition of the reactor, the emergency power supply requirement for the operable charging pump, and the fact that the plant is procedurally required to be borated to the highest required boron concentration for Modes 3, 4, or 5 prior to entering Mode 4. This provides assurance that reactivity control will be maintained and stable while only one charging pump is available. The additional options for disabling the charging pump (provided in the bases for TS 4.4.9.3.5) will allow for faster restoration when needed while maintaining two distinct operator actions to prevent a second pump from being started. This provides added assurance that reactor vessel integrity will be maintained.

Procedures will minimize the amount of time that both charging pumps are operable and having at least one pump in pull-to-lock will ensure that the second pump does not inadvertently start and exceed the Appendix G analysis limits and thus, ensure reactor vessel integrity.

The TS bases for requiring RHR in Mode 5 is to remove decay heat and provide RCS circulation. Since the RCP can perform the RHR circulation function and the decay heat removal function is not required during heatup, the proposed change is consistent with the bases. Since this option is only allowed during heatup where decay heat is low, sufficient time will be available to re-establish RHR heat removal as required to mitigate a LOP in Mode 5. Further, with the RCS pressurized, four RCS loops operable and the SG filled, core cooling can be accomplished by the steam generators via natural circulation once the plant heats up to Mode 4, in the event that RHR cannot be re-established. Therefore, the design basis analyses remain limiting and the margin of safety is not reduced.

The original plant design allows the RHR pumps to be available for both heat removal while shutdown and ECCS. As such, an allowance, TS 3.5.3.f, was provided to allow manual realignment from heat removal to ECCS mode. The specific wording of TS 3.5.3.f implies that this realignment only involves the suction valves. Since discharge valves must also be realigned, the TS is being reworded to apply for the discharge as well as suction valves. Therefore, this change is a clarification of the existing TS.

The proposed changes do not impact the protective boundaries (reactor vessel integrity) nor any of the design basis accidents.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: July 25, 1997

Description of amendment request: The proposed amendment request would implement 10 CFR Part 50 Appendix J, Option B by revising the Technical Specifications (TS) to allow the frequency of conducting integrated leak rate testing (ILRT) and local leak rate testing (Type B and C) to be based on component performance.

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 implements Option B of 10 CFR Part 50 Appendix J on performance-based containment leakage testing. The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameters or conditions that contribute to the initiation of any accidents previously evaluated. The proposed change potentially affects the leak-tight integrity of the containment structure designed to mitigate the consequences of a Loss-of-Coolant Accident (LOCA). The function of the containment is to maintain functional integrity during and following the peak transient pressures and temperatures and limit fission product leakage following the design basis LOCA. Because the proposed change does not alter the plant design, only the frequency of measuring Type A, B, and C leakage, the proposed change does not directly result in an increase in containment leakage.

Test intervals will be established based on the performance history of components being tested. The frequency of monitoring the relatively few containment isolation valves and/or containment penetrations subject to above normal leakage will not decrease by implementing Option B of Appendix J. A performance based program will identify those valves and penetrations which must continue to be tested each refueling outage.

The risk resulting from the proposed changes is characterized as follows, based primarily on the results contained in NUREG-1493 ''Performance-Based Containment Leakage Test Program,'' the principal Technical Support Document used by the NRC as the basis for the Appendix J Final Rule:

Type A Testing

NUREG-1493 found that the effect of containment leakage on overall accident risk is minimal since risk is dominated by accident sequences that result in failure or bypass of the containment. Industry wide, Integrated Leak Rate Tests (ILRTs) have only found a small fraction of the leaks that exceed current acceptance criteria. Only three percent of all leaks are detectable only by ILRTs, and therefore, by extending the Type A testing intervals, only three percent of all leaks have a potential for remaining undetected for longer periods of time. In addition, when leakage has been detected by ILRTs, the leakage rate has been only marginally above existing requirements. The Fort Calhoun Station Unit No. 1 Type A testing confirms the industry-wide experience that a majority of the leakage experienced during Type A testing is through components tested by Type B and C tests.

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

Type B and C Testing

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

Therefore, the proposed change will not result in 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.

There will be no physical alterations to the plant configuration, changes to setpoint values, or changes to the implementation of setpoints or limits as a result of this proposed change. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents.

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

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

The proposed change only affects the frequency of Type A, B, and C testing. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed.

The frequency of monitoring the relatively few containment isolation valves and/or containment penetrations subject to above normal leakage will not decrease by implementing Option B of Appendix J. A performance based program will identify those valves and penetrations which must continue to be tested each refueling outage. NUREG-1493 has determined that, under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to La, which is stated in the Fort Calhoun Station Unit No. 1 Technical Specifications to be 0.1 percent by weight of the containment air per 24 hours at 60 psig.

The limitation on containment leakage rate is designed to ensure that total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. The margin to safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 La acceptance criteria. The La value is not being modified by this proposed change.

Except for the method of defining the test frequency, no change in the method of testing is being proposed. The Type B and C tests will continue to be done at 60 psig or greater. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

Therefore, the proposed change will not result in a significant reduction in a margin of safety.

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

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William H. Bateman

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

Date of amendment request: August 26, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 4.6.5.3.1b, for the Filtration, Recirculation and Ventilation System (FRVS), Ventilation Subsystem, and TS 4.6.5.3.2b for the FRVS Recirculation Subsystem. The revised TSs would state that the heaters should be ''operating (automatic heater modulation to maintain relative humidity)'' instead of ''on'' when performing the 10-hour, monthly test.

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

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

The proposed TS revisions involve no hardware changes and no changes to existing structures, systems or components. Conducting TS Surveillance Requirements 4.6.5.3.1.b and 4.6.5.3.2.b with the FRVS recirculation unit and ventilation unit heaters in automatic modulation to maintain the relative humidity within the design requirements, meets the intent of the USNRC Regulatory Guide 1.52, position C.4.d, in reducing adsorber and HEPA filter moisture levels. In the unlikely event that the adsorber and HEPA filters, that are enclosed and isolated in a confined space should reach an equilibrium at the maximum design operating humidity level, the 10 hour run with heaters energized would reduce the humidity to acceptable levels. Therefore, the proposed changes do not change the post-accident performance characteristics of the FRVS adsorber or HEPA filters below the design requirements and does not increase the consequences of accidents previously identified. Since there are no changes to the operation of FRVS in normal or post-accident operating conditions, there is no increase in the probability of an accident previously evaluated.

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

The proposed changes contained in this submittal will not adversely impact the operation of any safety related component or equipment. PSE&G has concluded that [the] method of performing the monthly FRVS recirculation unit and ventilation unit surveillances with the heaters modulating adequately maintains and demonstrates operability of FRVS. Since the proposed changes involve: 1) no hardware changes; 2) no changes to FRVS operation in normal operating or post-accident conditions; and 3) no changes to existing structures, systems or components, there can be no impact on the potential occurrence of any accident. Furthermore, there is no change in plant testing proposed in this change request which could initiate an event. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The revisions to TS Surveillance Requirements 4.6.5.3.1.b and 4.6.5.3.2.b provide a more accurately defined basis for performing this surveillance test. The proposed changes reflect PSE&G's position on satisfying USNRC Regulatory Guide 1.52, position C.4.d. Since PSE&G has concluded that performing TS Surveillance Requirements 4.6.5.3.1.b and 4.6.5.3.2.b with the FRVS recirculation unit and ventilation unit heaters in automatic moduation [sic] [modulation] to maintain the relative humidity within the design requirements, adequately reduces adsorber and HEPA filter moisture levels, the proposed changes do not significantly reduce a margin of safety in FRVS. Since the FRVS recirculation units and ventilation units will continue to be tested with the heaters: 1) operable; and 2) set at the demand necessary to ''reduce the buildup of moisture,'' PSE&G believes that the proposed changes to clarify the TS are justified.

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

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit - N21, P.O. Box 236, Hancocks Bridge, NJ 08038

NRC Project Director: John F. Stolz

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

Date of amendment request: August 19, 1997

Description of amendment request: The proposed amendment would revise the Ginna Station Improved Technical Specifications (ITS) by revising the Emergency Core Cooling System Accumulators Surveillance Requirement 3.5.1.2 to correct the specified accumulator borated water volume values in order to match the associated accumulator percent level values.

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

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The change is only to correct a conversion error with respect to accumulator borated water volume. This does not increase the probability of any accident previously evaluated since the accumulator water volume provides mitigation capability only (i.e., does not initiate any accident). The affected accident analyses with respect to the accumulator (e.g., small and large [loss-of-coolant] LOCA and steam line break) have been re-evaluated using the correct accumulator water volume values with acceptable results. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. Ginna Station operators verify accumulator water volume via percent level (versus cubic feet) which remains unchanged. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only correct a conversion error. The error has been re-evaluated with acceptable results. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Alexander W. Dromerick, Acting Director

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

Date of amendments request: February 14, 1997, as supplemented by letters dated June 20, August 5, and September 22, 1997

Description of amendments request: The proposed amendments would change the maximum reactor core power level for facility operation from 2652 megawatts thermal (MWt) to 2775 MWt in the Farley, Units 1 and 2, Facility Operating Licenses. In addition, the proposed amendments would involve the following Technical Specification (TS) changes.

The defined rated thermal power for Farley; departure from nucleate boiling (DNB) parameters for reactor coolant system (RCS) average temperature (T_{avg}); pressurizer pressure; and RCS flow would be changed.

The reactor trip system interlock setpoint for power range neutron flux (P-8) and engineered safety features (ESF) actuation trip setpoint for steam generator water high-high level for turbine trip and feedwater isolation (P-14), and ESF actuation system interlock for low-low T_{avg} (P-12) would be modified to reflect analytical results.

An evaluation of additional reactor trip system and ESF actuation system safety analysis limits and trip setpoints would result in changes to the allowable values for several functions.

On the basis of the results of new containment analyses, the maximum peak calculated containment internal pressure for a loss-of-coolant accident (LOCA) event would be revised. The main steamline isolation valve closure time requirement would be revised. Surveillance requirements for emergency core cooling systems (ECCS) would be modified to reflect reduced ECCS flows. The number of secondary system hydrostatic pressure tests (Table 5.7-1) would be increased. For Farley Unit 2 only, the steam generator F* distance would be revised.

Changes to the plant design features and administrative controls are also proposed. These changes would revise the RCS fluid volume contained in Section 5.4 and the addition of the NRC-approved references for best estimate LOCA listed in Section 6.9.1.

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

DEFINITION, DESIGN FEATURE AND ADMINISTRATIVE CONTROL CHANGES

* * * *

1. The proposed changes to the rated thermal power definition, RCS fluid volume, and COLR [Core Operating Limit Report] references do not increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The comprehensive analytical efforts performed to support the proposed uprating included a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. The revised power uprate value and RCS fluid volume were inputs to applicable safety analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable. None of these proposed changes directly initiate any accident; therefore, the probability of an accident has not increased. All dose consequences have been analyzed or evaluated with respect to these parameters, and all acceptance criteria continue to be met. Therefore, the consequences of an accident previously evaluated in the FSAR have not increased.

2. The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed technical specification changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed operating license and technical specification changes do not involve a significant reduction in a margin of safety. All analyses supporting the proposed power uprate reflect the RCS fluid volume and rated thermal power values. The use of NRC approved BELOCA [best estimate LOCA] methodology must be referenced since BELOCA will now be the LBLOCA [large break LOCA] analysis licensing basis for FNP [Farley Nuclear Plant]. All acceptance criteria (including LOCA peak clad temperature, DNB criteria, containment temperature and pressure, and dose limits) continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

DNB PARAMETERS CHANGES

* * * *

1. The proposed technical specification changes for DNB parameters do not involve a significant increase in the probability or consequences of an accident previously evaluated in the FNP FSAR. The mechanical design features associated with VANTAGE 5 fuel and the improved methodologies (such as Revised Thermal Design Procedure) provide capability for relaxation of analytical input parameters such that increased DNBR [DNB ratio] margin can be generated without violation of any acceptance criteria. The indicated DNB parameters bound the analytical values used to support the proposed uprating. In each case, the appropriate design and acceptance criteria are met. All performance requirements for any system or component have been evaluated and support the revised analysis assumptions. Overall plant integrity is not reduced. Furthermore, the parameter changes are associated with features used as limits or mitigators to assumed accident scenarios and are not accident initiators. Therefore, the probability of an accident has not significantly increased.

The radiological consequences of accidents previously evaluated in the FSAR have been assessed due to the proposed technical specification changes. Evaluations have confirmed that the doses remain within previously approved acceptable limits as well as those defined by 10 CFR [Part] 100. Therefore, the radiological consequences to the public resulting from any accident previously evaluated in the FSAR has not significantly increased.

2. The proposed technical specification changes do not create the possibility of a new or different kind of accident from any previously evaluated in the FSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the revised DNB parameters. The revised analytical assumptions have no adverse effect and do not challenge the performance of any other safety-related system. This has been verified in WCAP 12771, Rev. 1. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed technical specification changes do not involve a significant reduction in the margin of safety. The margin of safety for fuel-related parameters (such as DNB and Kw/ft) are defined in the Bases to the Technical Specifications. The uncertainties associated with the proposed DNB parameter changes are included in the core safety limits. Performance of analyses and evaluations with the reactor core safety limits defined by RTDP [Revised Thermal Design Procedure] have confirmed that the operating envelope defined by the Technical Specifications continues to be bounded by the revised analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided by the analyses in accordance with these acceptance limits is not reduced.

MISCELLANEOUS OPERATION AND MARGIN ENHANCEMENT CHANGES

* * * *

1. The proposed changes do not increase the probability or consequences of an accident previously evaluated in the FSAR. Explicit modeling of these parameters is included in the uprate analyses and evaluations. The comprehensive analytical effort performed to support the proposed uprating has included a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. In addition LOCA and non-LOCA analyses and evaluations have verified that all acceptance criteria continue to be met. All systems will function as designed. None of these proposed changes can directly initiate any accidents; therefore, the probability of an accident has not been increased. All dose consequences have been analyzed or evaluated with respect to these parameters, and all acceptance criteria continue to be met. Therefore, the consequences of an accident previously evaluated in the FSAR have not increased.

2. The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed technical specification changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed technical specification changes do not involve a significant reduction in a margin of safety. All analyses supporting the proposed power uprate reflect these proposed values. All acceptance criteria (including LOCA peak clad temperature, DNB criteria, containment temperature and pressure, and dose limits) continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

ALLOWABLE VALUES AND TRIP SETPOINTS FOR REACTOR TRIP SYSTEM AND ESFAS [ENGINEERED SAFETY FEATURE ACTUATION SYSTEM]

* * * *

1. The proposed changes do not increase the probability or consequences of an accident previously evaluated in the FSAR. The comprehensive engineering effort performed to support the proposed uprating has included evaluations or reanalysis of all accident analyses including all dose related events. Setpoint calculations have verified acceptability of the proposed setpoints and allowable value changes. All systems will function as designed, and all performance requirements on these systems have been verified to be acceptable. Neither allowable values nor the setpoints initiate any accident; therefore, the probability of an accident has not been increased. All dose consequences have been analyzed or evaluated with respect to these parameters, and all acceptance criteria continue to be met. Therefore the consequences of an accident previously evaluated in the FSAR have not increased.

2. The proposed setpoints and allowable value changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed technical specification changes have no adverse effects on any safety-related system and do not challenge the performance of integrity of any safety-related system. The specified trip setpoints associated with the respective RTS [Reactor Trip System] and ESFAS functions ensure all accident analyses criteria continue to be met. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed technical specification changes do not involve a significant reduction in a margin of safety. All analyses supporting the proposed power uprate reflect these proposed values. Setpoint calculations demonstrate that margin exists between the setpoint and the corresponding safety analysis limits. The calculations are based on FNP instrumentation and calibration/functional test methods and include allowances for uprated power conditions. All acceptance criteria (including LOCA peak clad temperature, DNB criteria, containment temperature and pressure, and dose limits) continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: September 17, 1997 (TS 97-02)

Description of amendment request: The proposed changes would revise Section 4.6.2.1 of the Sequoyah Technical Specifications (TS) to change the parameters to be monitored during the inservice inspection surveillance testing of the containment spray system pumps. The changes would also adopt provisions in the Westinghouse Improved Standard TS (NUREG-1431) that affect that section of the Sequoyah TS.

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

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

The proposed revisions to the containment spray system surveillances for the pumps, valves, and nozzles do not change the intent of the current TS requirements. These revisions only affect the TS operability testing requirements without changing the system functions. These functions are not considered to be accident initiators. The proposed surveillance wording is not based on changes to the plant although a modification to flow orifices for the containment spray pumps created the need to revise the surveillance that verifies pump developed head. The revisions primarily provide flexibility for required methods to verify system operability as well as utilizing less prescriptive operability limits and conditions for testing. The testing flexibility and less prescriptive requirements do not relax the intent to properly verify operability of the containment spray system but do allow for changes in testing that continue to ensure the appropriate operability requirements. Since these revisions are not directly related to modifications of the plant or result in different methods for operating the plant, there is no change that could increase the probability of an accident. In addition, the consequences of an accident are not increased because there has not been a change that would impact the safety functions of the containment spray system. These revisions will continue to properly verify the operability of the containment spray system.

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

The containment spray system functions are not changed as discussed above and the operating practices for the plant remain the same. The testing methods can be modified as a result of the proposed revisions but will continue to maintain appropriate verifications of system operability. These testing methods as well as the containment spray system are not considered to be a potential initiator of accidents. Therefore, these revisions will not impact the operation of systems that could initiate an accident and the possibility of a new or different kind of accident is not created.

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

The proposed revisions do not directly change the limits for containment spray system operability although they do provide the flexibility to properly revise limits resulting from system modifications. This type of limit revision would be necessary to adequately verify system operability. The appropriate limits continue to be required by the proposed TS surveillance requirements. Therefore, the proposed revisions do not allow inappropriate changes to setpoints or operating requirements that maintain the margin of safety and no reduction in this margin is involved in this request.

The NRC 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: August 26, 1997

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.2, ''Power Distribution Limits.'' The DNB Parameters Limiting Condition for Operation would be modified consistent with an industry notification.

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, assumptions or probabilities are affected by the proposed change. The proposed change corrects a nonconservative Technical Specification Action statement by removing provisions which allow continued Mode 1 plant operation in the event the Reactor Coolant System flow rate is less than the required value. Under the proposed change, a power reduction to less than 5 percent of rated thermal power (Mode 2) will be required if the Reactor Coolant System flow rate is less than the required Technical Specification value.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not affect any equipment, accident conditions, or assumptions which could lead to a significant increase in radiological consequences of an accident. The proposed change will ensure accident analyses remain valid if the Reactor Coolant System flow rate becomes less than the required value.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators will be introduced by the proposed change. No equipment or operations will be affected.

3. Not involve a significant reduction in a margin of safety because under the proposed Technical Specification Action statement a power reduction to less than 5 percent of rated thermal power (Mode 2) will be required if degraded Reactor Coolant System flow develops. The proposed Action statement ensures accident analyses' assumptions are maintained.

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

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

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

NRC Project Director: Gail H. Marcus

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: August 22, 1997, as supplemented by letter dated September 18, 1997

Description of amendment request: The proposed amendment would revise the Vermont Yankee Technical Specifications (TSs) to address the new low pressure $C0_2$ suppression system for the East and West Switchgear Rooms and more clearly describe the separation of the rooms.

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 changes support the use of a newly installed low pressure $C0_2$ suppression system for the East and West Switchgear Rooms, to meet the $C0_2$ concentration requirements of NFPA 12 (1993) following detection of a fire condition in one of the associated rooms. The new low pressure $C0_2$ system consists of a 6 ton storage tank, piping, valves, associated instrumentation and controls.

The FSAR [Final Safety Analysis Report] was reviewed for impact as a result of this proposed amendment with none being found. The initiators of the four design basis accidents, as defined in section 14.6 of the FSAR, were reviewed with respect to the new low pressure $C0_2$ system. The low pressure $C0_2$ system is not an initiator of any of the Chapter 14.6 accidents. The low pressure $C0^{T22}$ suppression system is classified as a Non Nuclear Safety (NNS) related system. However, the $C0_2$ dispersion headers have been seismically mounted to preclude the possibility of their failure affecting safety related equipment during a seismic event. Although the Switchgear Room (East and West) low pressure $C0_2$ system is not used as a mitigator of any accident listed in section 14.6 of the FSAR, the switchgear contained in the aforementioned rooms is used to mitigate the consequences of the section 14.6 accidents.

The new low pressure $C0_2$ system, which meets NFPA 12 (1993), provides fire suppression for the affected room by raising the $C0_2$ concentration to a 50% level and maintains this concentration for a 20 minute duration upon initiation. As a result, this $C0_2$ system prevents a fire in the affected room from spreading to adjacent rooms and adversely impacting the adjacent room's safety related equipment. Consequently, the unaffected rooms and associated trains of equipment remain functional to perform their intended safety functions if required. The proposed amendment also reflects the separation of the switchgear room into two fire areas with equivalent detection and suppression.

Based on the above, use of the low pressure $C0_2$ system for East or West Switchgear Room fire suppression does not create new initiators, nor degrade the effectiveness of equipment relied upon to perform mitigative functions assumed for the previously evaluated design basis accidents. Therefore, the proposed amendment will 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 previously evaluated:

The NNS low pressure $C0_2$ system, which meets NFPA 12 (1993), provides fire suppression for the East and West Switchgear Rooms by raising the $C0_2$ concentration to a 50% level and maintains this concentration for a 20 minute duration upon initiation. As a result, this $C0_2$ system prevents a fire in the affected switchgear rooms from spreading to adjacent rooms and adversely impacting the adjacent rooms associated equipment. The switchgear room is more clearly depicted as two separate fire areas in the proposed amendment with equivalent protection. The $C0_2$ suppression header piping located in the switchgear rooms is seismically supported, which precludes the possibility of this piping failing during a seismic event and affecting safety related equipment located nearby.

The new low pressure $C0_2$ system does not introduce new accident initiators. The low pressure $C0_2$ system is fulfilling the fire suppression function previously performed by the existing high pressure $C0_2$ system. The previous separation of the switchgear room into two separate fire areas, provides separation of redundant equipment and equivalent fire detection and suppression for that equipment. The low pressure $C0_2$ system consists of a 6 ton storage tank, piping, valves, and associated instrumentation and controls. There are no failure mechanisms, associated with the new low pressure $C0_2$ equipment, which cannot be categorized under at least one of the three failure mechanisms identified in section 14.4.3 of the FSAR. Consequently, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

Technical Specifications 3.13.D/4.13.D were reviewed with respect to the proposed amendment to determine if the changes would result in a reduction in a margin of safety. The proposed amendment, to allow use of a low pressure $C0_2$ suppression system for the East or West Switchgear Rooms, does not degrade the existing fire protection program. The level of protection provided by the switchgear room $C0_2$ fire protection system is enhanced by the introduction of the new low pressure system which meets NFPA 12 (1993) and provides fire suppression for the East or West Switchgear Rooms by raising the $C0_2$ concentration to a 50% level and maintains this concentration for a 20 minute duration upon initiation. Consequently, the pre-established levels of system operability in the event of a fire and the assurance of a safe reactor shutdown, as provided by the fire protection systems, have not been degraded. An analysis has been performed to ensure that either a failure of the low pressure $C0_2$ storage tank outside the switchgear rooms, or a continuous discharge of the entire tank contents within the switchgear room, will not adversely affect either control room habitability or emergency diesel operation. The designation of separate fire areas for the switchgear room with equivalent protection does not decrease safety for this equipment. As a result, the proposed amendment will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, 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: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Ronald B. Eaton, Acting Project Director

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

Date of amendment request: July 16, 1997

Description of amendment request: The proposed amendment would add new minimum reactor vessel pressure versus reactor vessel metal temperature (P/T) curves, applicable to 12 EFPY (effective full power years). These changes are necessary to support leak and hydrostatic testing in accordance with the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Section XI.

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 12 EFPY curve was developed using the same methodology as that used in the current 32 EFPY curve and the 8 EFPY curve. This methodology is consistent with the guidance provided in Regulatory Guide 1.99, Revision 2.

Assumptions and parameters were the same as those used in the 8 EFPY curve calculation. However, fluence values used in the calculation were those for 12 EFPY.

Use of the 12 EFPY curves on or before attainment of 12 EFPY of operation is equivalent to the previously approved use of the 32 EFPY curves on or before attainment of 32 EFPY of operation.

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

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

The proposed change introduces no credible mechanism for unacceptable radiation release.

The proposed change does not require physical modification to the plant.

The 12 EFPY curves are consistent with the previously approved 32 and 8 EFPY curves.

Inservice hydrostatic or leak testing is not assumed to be an initiator of analyzed events. Since approval of the proposed amendment will ensure adequate protection of the reactor pressure vessel, it will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The accident analyses for the plant as described in the FSAR are not affected by this proposed change.

The 12 EFPY curves were developed using the same methodology as the 32 and 8 EFPY curves and thus involve no reduction in the margin of safety as previously evaluated.

The margin of safety, relative to the available heat sink in the Reactor Coolant System, is actually increased by the use of the proposed curves due to the lower allowed test temperature.

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 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: Richland Public Library, 955 Northgate Street, Richland, Washington 99352

Attorney for licensee: Perry D. Robinson, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

NRC Project Director: William H. Bateman

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

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

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

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: August 6, 1997, as supplemented August 26, 1997

Brief description of amendments: The proposed amendments would address an unreviewed safety question associated with handling of the spent fuel shipping cask at the Brunswick Steam Electric Plant, Units 1 and 2.Date of publication of individual notice in **Federal Register:** September 17, 1997 (62 FR 48897)

Expiration date of individual notice: October 17, 1997

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

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 application for amendment: August 26, 1997

Brief description of amendment request: The proposed amendments would approve a modification to the Diablo Canyon Power Plant, Unit Nos. 1 and 2 auxiliary saltwater (ASW) system to bypass approximately 800 feet of Unit 1 and 200 feet of Unit 2 Class 1 ASW pipe, a portion of which is buried below sea level in the tidal zone outside the intake structure. This modification was completed on Unit 1 during the refueling outage completed this year.Date of individual notice in **Federal Register:** September 16, 1997 (62 FR 48677)

Expiration date of individual notice: October 16, 1997

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

Southern Nuclear Operating Company, Inc., et al., Docket No. 50-348, Joseph M. Farley Nuclear Plant, Unit No. 1, Houston County, Alabama

Date of amendment request: September 3, 1997

Description of amendment request: The proposed amendment would allow a reduction in the number of required available movable detector thimbles (flux map paths) for Cycle 15 operation.Date of publication of individual notice in Federal Register: September 10, 1997 (62 FR 47695)

Expiration date of individual notice: October 10, 1997

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

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

Date of amendment request: September 17, 1997

Description of amendment request: The proposed amendments would modify Technical Specification 3/4.4.9, ''Specific Activity,'' and associated Bases to reduce the limit associated with dose equivalent iodine-131.Date of publication of individual notice in Federal Register: September 24, 1997 (62 FR 49998)

Expiration date of individual notice: October 24, 1997

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

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

Date of application for amendment: August 14, 1997

Brief description of amendment: The proposed amendment would change the Perry Nuclear Power Plant design basis as described in the Updated Safety Analysis Report. The change will add a description of the methodology utilized for determining the systems and components that are considered to require protection from tornado missiles.Date of individual notice in **Federal Register:** September 16, 1997 (62 FR 48674).

Expiration date of individual notice: October 16, 1997

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: December 27, 1996, as supplemented by letter dated August 22, 1997

Brief description of amendment: The amendments change Technical Specification 3/4.6.1.3.b and its associated Bases sections to reflect an increase in the peak containment internal pressure for the design basis loss-of-coolant accident (LOCA) from 49.5 psig to 52 psig.

Date of issuance: September 11, 1997

Effective date: September 11, 1997, to be implemented within 30 days from its date of issuance.

Amendment No.: Unit 1 - 113; Unit 2 - 106; Unit 3 - 85

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27794) The August 22, 1997, supplemental letter provided additional clarifying information and did not change the staff's original no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 8, 1997, as supplemented August 22, 1997

Brief description of amendments: These amendments remove the suppression chamber water volume band from Technical Specification 3.6.2.1.a.1 while retaining the equivalent water level band. The amendments additionally revised the volume band to account for the displacement of water due to the installation of larger emergency core cooling system suction strainers.

Date of issuance: September 17, 1997

Effective date: September 17, 1997

Amendment Nos.: 188 and 219

Facility Operating License Nos. DPR-71 and DPR-62: Amendments change the Technical Specifications

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43366) The August 22, 1997, submittal provided a correction to the Bases to reflect a change authorized by a previous amendment and did not alter the initial no significant hazards determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 17, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: July 1, 1997

Brief description of amendments: The amendments revise Technical Specification definition 1.4, Channel Calibration, to allow an alternative method of calibrating thermocouples and resistance temperature detector sensors. The amendments also make editorial and administrative corrections to TS Table 3.3.2-1, Table 3.3.6-1, and Bases Section 3/4.3.1.

Date of issuance: September 15, 1997

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

Amendment Nos.: 102 and 104

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40848) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 15, 1997. No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: October 30, 1996, as supplemented by letters dated April 22, July 2, September 3, and September 4, 1997

Brief description of amendments: The amendments revise the Reactor Building Structural Integrity Technical Specifications regarding the tendon surveillance program.

Date of Issuance: September 15, 1997

Effective date: The license amendments are effective as of the date of issuance and the change to the facilities shall be implemented prior to the Unit 1 end-of-cycle 17 outage. Implementation of the amendments shall include the provisions that the licensee provide in the facility Updated Final Safety Analysis Report (specifically the Selected Licensee Commitment Manual) the prescribed lower limit and the minimum required value of Reactor Building Post-Tensioning System tendon forces for each group of tendons prior to performing the seventh tendon surveillance for Unit 1. In addition, the portion of the Selected Licensee Commitment Manual related to the establishment of these limits will be submitted as soon as available.

Amendment Nos.: 225, 225, 222

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the Technical Specifications, License Conditions, and Appendix C.

Date of initial notice in **Federal Register:** December 4, 1996 (61 FR 64383) The April 22, July 2, September 3, and September 4, 1997, letters provided clarifying information that did not change the scope of the October 30, 1996, application and the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 15, 1997.No significant hazards consideration comments received: No.

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

Duke Energy Corporation, Docket Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: June 12, 1997

Brief description of amendments: The amendments change the name ''Duke Power Company'' to ''Duke Energy Corporation'' in the Oconee facility operating licenses and Technical Specifications.

Date of Issuance: September 16, 1997

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

Amendment Nos.: 226, 226, 223

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the Technical Specifications and Operating Licenses including Appendix C.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35849) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 16, 1997, and Environmental Assessment dated August 21, 1997 (62 FR 44495).No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 6, 1996, as supplemented May 23, 1997 and August 13, 1997.

Brief description of amendments: These amendments revise Item 7.c of Beaver Valley Power Station, Unit No. 1 (BVPS-1) Technical Specification (TS) Table 3.3-3 and Item 7.d of Beaver Valley Power Station, Unit No. 2 (BVPS-2) TS Table 3.3-3 to reflect that a safety injection (SI) signal starts all auxiliary feedwater (AFW) pumps. The notation on BVPS-1 TS Table 3.3-5 is revised to state that the response time is for all AFW pumps on all SI signal starts. Items 7.d of BVPS-2 TS Tables 3.3-4 and 4.3-2 is revised to reflect that an SI signal starts all AFW pumps.

The amendments also revise and reformat TSs 3/4.7.1.2 to more closely resemble the wording contained in the NRC's ''Standard Technical Specifications Westinghouse Plant,'' (NUREG-1431, Revision 1). These changes require three AFW trains to be operable and describe what constitutes an operable train. The mode applicability for these TSs is expanded to include Mode 4 when the steam generator(s) is relied upon for heat removal.

Date of issuance: September 18, 1997

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

Amendment Nos.: 206 and 85

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

Date of initial notice in **Federal Register:** November 19, 1996 (61 FR 58902) The May 23, 1997, and August 13, 1997, letters provided minor editorial changes that did not change the initial proposed no significant hazards consideration determination or expand the amendment request beyond the scope of the November 19, 1996, Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 18, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 29, 1997

Brief description of amendment: The amendments consist of changes to the Technical Specifications (TS) which correct typographical errors, remove outdated material, incorporate minor changes in text, make editorial corrections, and resolve other inconsistencies in the Unit 1 and 2 TS.

Date of Issuance: September 22, 1997

Effective Date: September 22, 1997

Amendment Nos.: 152 and 89

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40849) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1997.No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Community College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34981-5596

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

Date of amendment request: July 8, 1996

Brief description of amendments: The amendments allowed that the component cooling water system surge tank level instrumentation can be demonstrated operable, by performing a channel calibration test, during any plant mode of operation.Date of issuance: September 23, 1997

Effective date: September 23, 1997, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 91; Unit 2 - Amendment No. 78

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

Date of initial notice in **Federal Register:** August 28, 1996 (61 FR 44358) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 23, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 19, 1997

Brief description of amendment: Technical Specification 3/4.7.1.3 requires sufficient water to be available for the auxiliary feedwater system to maintain the reactor coolant system at hot standby for 10 hours before cooling down to hot shutdown in the next 6 hours. The amendment increases the required volume of water when the demineralizer water storage tank and condensate storage tank are being credited, makes editorial changes, and expands the descriptions in Bases Sections 3/4.7.1.2 and 3/4.7.1.3.

Date of issuance: September 11, 1997

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

Amendment No.: 150

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40853) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 7, 1997, as supplemented May 30, July 29, and September 12, 1997

Brief description of amendments: The amendments revise Technical Specification (TS) 3.8, including TS 3.8.D.1 and TS 3.8.D.3, to change TS limitations on crane operations in the spent fuel pool enclosure relating to spent fuel pool special ventilation system operability. These changes are necessary to allow movement of loads over spent fuel stored in the spent fuel pool enclosure with the spent fuel pool special ventilation system inoperable. The staff denied the proposed change to TS 3.8.D.2. A separate notice of denial has been sent to the **Federal Register** for publication.

Date of issuance: September 15, 1997

Effective date: September 15, 1997, with full implementation within 30 days. License Condition 4 of Appendix B is effective immediately upon issuance of the amendments.

Amendment Nos.: 130 and 122

Facility Operating License Nos. DPR-42 and DPR-60: Amendments revised the Licenses and Technical Specifications.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35850) The July 29 and September 12, 1997, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 15, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 3, 1997

Brief description of amendment: This amendment makes changes to Technical Specification Table 3.6.3-1, ''Primary Containment Isolation Valves'' to add valves to the list, therein.

Date of issuance: September 15, 1997

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

Amendment No.: 102

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43375) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 7, 1997

Brief description of amendment: The amendment changes Technical Specification (TS) 3/4.8.4.2, ''Motor Operated Valves - Thermal Overload Protection (BYPASSED),'' to relocate the list of applicable valves (TS Table 3.8.4.2-1) to the Hope Creek Generating Station Updated Final Safety Analysis Report.

Date of issuance: September 16, 1997

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

Amendment No.: 103

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43375) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 16, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 1, 1997, as supplemented by letter dated May 30, 1997

Brief description of amendment: The amendment changed Technical Specifications (TSs) 4.6.1.1, ''Primary Containment Integrity;'' 3/4.6.1.2, ''Primary Containment Leakage;'' 3/4.6.1.3, ''Primary Containment Air Locks;'' 4.6.1.5.1, ''Primary Containment Structural Integrity;'' and 4.6.1.8.2, ''Drywell and Suppression Chamber Purge System.'' This amendment also changed the Bases for 3/4.6.1.2, ''Primary Containment Leakage;'' 3/4.6.1.3, ''Primary Containment Air Locks;'' 3.4.6.1.5, ''Primary Containment Structural Integrity;'' Section 6, ''Administrative Controls;'' and License Condition 2.D of Facility Operating License NPF-57. A new TS, 6.8.4.f, ''Primary Containment Leakage Rate Testing Program,'' was added. These changes modify the TSs and the Facility Operating License to adopt the performance based containment leak rate testing requirements (Option B) of 10 CFR Part 50, Appendix J.Date of issuance: September 18, 1997

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

Amendment No.: 104

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43375) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 18, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 26, 1997

Brief description of amendment: The amendment changes the definition of ''Core Alteration.''

Date of issuance: September 17, 1997

Effective date: September 17, 1997

Amendment No.: 138

Facility Operating License No. NPF-12: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27800) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 17, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendments: September 26, 1996, as supplemented on August 12, 1997 (TS 96-04)

Brief description of amendments: The amendments change the Technical

Specifications (TS) by relocating the fire protection program details to the Updated Final Safety Analysis Report and Fire Protection Plan in accordance with Generic Letters 86-10 and 88-12.

Date of issuance: September 23, 1997

Effective date: September 23, 1997

Amendment Nos.: 228 and 219

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the TS.

Date of initial notice in **Federal Register:** July 2, 1997 (62 FR 35843)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 23, 1997.No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket No. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: April 30, 1997, as supplemented June 18, July 21 (3 letters), August 7 and 21, 1997

Brief description of amendment: The proposed amendment would change the design features section of the Technical Specifications (TS) to provide for insertion of Lead Test Assemblies containing Tritium Producing Burnable Absorber Rods in the Watts Bar Nuclear Plant reactor core during Cycle 2.

Date of issuance: September 15, 1997

Effective date: September 15, 1997

Amendment No.: 8

Facility Operating License No. NPF-90: Amendment revises the TS.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30644) The TVA letters dated June 18, July 21, August 7 and 21, 1997 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 an Environmental Assessment dated September 8, 1997, and in a Safety Evaluation dated September 15, 1997.No significant hazards consideration comments received: None.

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

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

Date of application for amendment: January 16, 1996, supplemented December 6, 1996, and August 15, 1997

Brief description of amendment: The amendment extended the test interval for the drywell bypass leakage rate test from 18 months to 10 years. Also, some surveillances for the drywell air locks were increased from 18 months to 24 months.

Date of issuance: September 22, 1997

Effective date: September 22, 1997

Amendment No.: 88

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

Date of initial notice in **Federal Register:** February 2, 1996 (61 FR 3951) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 2, 1997

Brief description of amendment: The amendment revises an existing exception to Limiting Condition for Operation (LCO) 3.0.4 as it applies to LCO 3.6.1.9 for the main steam isolation valve (MSIV) leakage control system (LCS) by making the exception permanent and clarifying that it only applies for the inboard MSIV LCS subsystem.

Date of issuance: September 24, 1997

Effective date: September 24, 1997

Amendment No.: 89

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33135) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 1997.No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 14, 1997

Brief description of amendment: The amendment revises Technical Specification (TS) 5.5.6 by adding a note that would extend the surveillance interval to perform the inservive testing (IST) full stroke exercise of primary containment isolation check valve TIP-V-6 until the 1998 refueling outage, scheduled to begin no later than May 15, 1998, or until a plant shutdown of sufficient duration occurs to allow TIP-V-6 testing, whichever occurs first.

Date of Issuance: September 18, 1997

Effective date: September 18, 1997, to be implemented within 30 days of issuance.

Amendment No.: 152

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 45280 dated August 26, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by September 25, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation and final no significant hazards consideration determination are contained in a Safety Evaluation dated September 18, 1997.

Attorney for licensee: Perry D. Robinson, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

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

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

Date of application for amendments: January 16, 1997 (TSCR-191), as supplemented on April 17, August 7, and August 27, 1997

Brief description of amendments: These amendments increase the minimum volume and boron concentration for the refueling water storage tanks and the boric acid storage tanks. Additionally, these amendments increase the minimum concentration of boric acid in the safety injection accumulator, the reactor coolant system during refueling operations, and the reactor coolant system during positive reactivity changes made when containment integrity is not maintained.

Date of issuance: September 23, 1997

Effective date: September 23, 1997, with full implementation within 45 days

Amendment Nos.: 180 and 184

Facility Operating License Nos. DPR-24 and DPR-27: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 23, 1997 (62 FR 19836) The April 17, August 7, and August 27, 1997, submittals provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 23, 1997.No significant hazards consideration comments received: No.

Local Public Document Room location: The Lester Public Library, 1001 Adams Street, Two Rivers, Wisconsin 54241

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

Date of amendment request: July 3, 1997

Brief description of amendment: The amendment changes the definition for an alteration of the reactor core to one that is consistent with the intent of the Improved Standard Technical Specifications.

Date of issuance: September 18, 1997

Effective date: September 18, 1997

Amendment No.: 109

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40861) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 18, 1997.No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 3, 1997

Brief description of amendment: The amendment modifies Technical Specifications 5.3.1, ''Fuel Assemblies'' and 6.1.9.6, ''CORE OPERATING LIMITS REPORT (COLR)'' to add ZIRLO as fuel material and the use of limited zirconium alloy filler rods in place of fuel rods.

Date of issuance: September 22, 1997

Effective date: September 22, 1997, to be implemented within 30 days of issuance.

Amendment No.: 110

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40860) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1997.No significant hazards consideration comments received: No.

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

Notice Of Issuance Of Amendments To Facility Operating LicensesAnd Final Determination Of No Significant Hazards ConsiderationAnd Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 7, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Pennsylvania Power and Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: September 15, 1997, as supplemented by letter dated September 16, 1997

Brief description of amendment: The amendment revised the applicability requirement in Technical Specifications (TSs) Sections 3.4.2, ''Safety/Relief Valves'' (Action c), 4.4.2, and 3.3.7.5, ''Accident Monitoring Instrumentation'' (TS Table 3.3.7.5-1, Action 80). The change to the referenced TSs adds the following applicability footnote: Compliance with these requirements for the ''S'' SRV acoustic monitor is not required for the period beginning September 12, 1997, until the next unit shutdown of sufficient duration to allow for containment entry, not to exceed the 10th refueling and inspection outage.

Date of issuance: September 23, 1997

Effective date: September 23, 1997

Amendment No.: 169

Facility Operating License No. NPF-14: This amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No. On September 17, 1997, the staff issued a Notice of Enforcement Discretion, which was immediately effective and remained in effect until this amendment was issued.

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

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

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

NRC Project Director: John F. Stolz

Dated at Rockville, Maryland, this 1st day of October 1997.

For the Nuclear Regulatory Commission

John N. Hannon,

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

[Doc. 97–26502 Filed 10–7–97; 8:45 am]

BILLING CODE 7590–01–F

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is a proposed Revision 1 to Regulatory Guide 3.54, and it is temporarily identified as DG–3010, ''Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation.'' The guide is in Division 3, ''Fuels and Materials Facilities.'' This regulatory guide is being revised to present a method that is acceptable to the NRC staff for calculating heat generation rates for use as design input for an independent spent fuel storage installation. The procedures proposed in this guide, for both boiling water reactors and pressurized water reactors, are simpler and therefore are expected to be more useful to applicants and reviewers.

The draft guide has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the guide. Comments should be accompanied by supporting data. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by January 2, 1998.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415–5905; e-mail CAG@nrc.gov. The draft guide may also be viewed and downloaded at this interactive site.

Regulatory guides are available for inspection at the Commission's Public Document Room, 2120 L Street NW., Washington, DC. Requests for single copies of draft or final guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Printing, Graphics and Distribution Branch; or by fax at (301) 415–5272. Telephone requests cannot be accommodated. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 12th day of September 1997.

For the Nuclear Regulatory Commission.

Joseph A. Murphy,

Director, Division of Regulatory Applications, Office of Nuclear Regulatory Research.

[FR Doc. 97–26639 Filed 10–7–97; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application to Withdraw From Listing and Registration; (Anicom, Inc., Common Stock, $.001 Par Value) File No. 1–13642

October 1, 1997.

Anicom, Inc. (''Company'') has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') and Rule 12d2–2(d) promulgated thereunder, to withdraw the above specified security (''Security'') from listing and registration on the Chicago Stock Exchange, Inc. (''CHX'' or ''Exchange'').

The reasons cited in the application for withdrawing the Security from listing and registration include the following:

The Board of Directors of the Company believes that maintenance of the dual listing on both the Chicago Stock Exchange and the Nasdaq National Market is not in the best interests of the Company's stockholders. No trading of the Company's Security has taken place on the CHX since May 1995. All of the trading activity in the Security has taken place on the Nasdaq National Market. Accordingly, the Board of Directors believes that the costs of maintaining a listing on the CHX do not justify the Company's continued listing on the CHX given the lack of trading on the Exchange.

By letter dated September 10, 1997, the CHX states that the Company has complied with the Exchange's rules relating to the delisting of the Company's Security.

Any interested person may, on or before October 23, 1997, submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW., Washington, DC 20549, facts bearing upon whether the application has been made in accordance with the rules of the exchange and what terms, if any, should be imposed by the Commission for the protection of investors. The Commission, based on the information submitted to it, will issue an order granting the application after the date mentioned above, unless the Commission determines to order a hearing on the matter.

For the Commission, by the Division of Market Regulation, pursuant to delegated authority.

Jonathan G. Katz,

Secretary.

[FR Doc. 97–26572 Filed 10–7–97; 8:45 am]

BILLING CODE 8010–01–M

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application to Withdraw From Listing and Registration; (Apollo Eye Group, Common Stock, $0.001 Par Value and Redeemable Warrants) File No. 1–12812

October 1, 1997.

Apollo Eye Group (''Company'') has filed an application with Securities and Exchange Commission (''Commission''),