only; no specific action or written response is required. Conformance with the guidance provided in the generic letter is voluntary.

The generic letter is available in the NRC Public Document Room under accession number 9711050091.

DATES: The generic letter was issued on November 13, 1997.

ADDRESSEES: Not applicable.

FOR FURTHER INFORMATION CONTACT: Beth A. Wetzel at (301) 415–1355.

SUPPLEMENTARY INFORMATION: Addressees of GL 96–06 have experienced difficulty in determining and implementing corrective actions for resolving the issues identified in the generic letter. Additionally, questions have been raised regarding (1) the risk implications of installing relief valves to deal with the thermal overpressurization issue; (2) the use of the ASME Code, Section III, Appendix F, criteria for permanent resolution of the thermal overpressurization issue; and (3) the NRC staff's closure of Generic Safety Issue 150, ''Overpressurization of Containment Penetrations.'' Given these considerations, risk insights, and industry initiatives that are being considered or that may be proposed, addressees may require additional time to fully evaluate and resolve the GL 96–06 issues. Therefore, addressees who find it necessary to revise their corrective actions or schedular commitments for resolving GL 96–06 issues may submit a revised response to the generic letter. Nevertheless, specific corrective actions that have been defined and are clearly needed should not be delayed without suitable justification.

Dated at Rockville, Maryland, this 13th day of November 1997.

For the Nuclear Regulatory Commission.

David B. Matthews,

Acting Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 97–30332 Filed 11–18–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATE: Weeks of November 17, 24, December 1 and 8, 1997.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of November 17

Friday, November 21

11:30 a.m. Affirmation Session (Public Meeting)

A. Final Rule—Deliberate Misconduct by Unlicensed Persons (Tentative)

B. Louisiana Energy Services—Financial Qualifications Aspects of Petitions for Review of LBP–96–25 (Contact: Ken Hart, 301–415–1659)

Week of November 24—Tentative

There are no meetings the week of November 24.

Week of December 1—Tentative

There are no meetings the week of December 1.

Week of December 8—Tentative

Thursday, December 11

2:00 p.m. Briefing on Investigative Matters (Closed—Ex. 5 & 7)

3:00 p.m. Affirmation Session (Public meeting) (if needed)

Friday, December 12

9:00 a.m. Meeting with Northeast Nuclear on Millstone (Public meeting) (Contact: Bill Travers, 301–415–1200)

Note: The schedule for commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

The NRC Commission Meeting Schedule can be found on the Internet at: http://nrc.gov/SECY/smj/schedule.htm.

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661).

In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: November 14, 1997.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 97–30527 Filed 11–17–97; 12:32 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 27, 1997, through November 6, 1997. The last biweekly notice was published on November 5, 1997 (62 FR 59912).

Notice Of Consideration Of Issuance Of Amendments ToFacility 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 December 19, 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.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: October 22, 1997

Description of amendments request: The proposed amendment incorporates both steady state and transient degraded voltage setpoints into Technical Specifications, as opposed to the current single degraded voltage setpoint. The proposed changes ensure adequate terminal voltage to all safety-related equipment during steady state and transient voltage conditions. Additionally, the 4 kV voltage range required during testing of the emergency diesel generators (EDGs) will be decreased to ensure the new steady state degraded voltage relays are not actuated during testing and to ensure the 4 kV motors are operated within their voltage rating.

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

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

The proposed changes revise the current degraded voltage setpoint and adds an additional steady state undervoltage requirement to Unit 1 and 2 Technical Specifications. The current degraded voltage relays will be referred to as

≥transient degraded voltage relays.'' The new settings allow for calibration tolerances, potential transformer correction factors, test equipment uncertainties, and relay drift. The nominal settings account for the above factors, plus additional margin to the analytical limit. The acceptable voltage range during EDG surveillance testing is also being decreased. The setpoint and time delay associated with the 4 kV bus loss of voltage relays is unaffected by this amendment request.

The accident analyses credit the loading of the EDGs based on loss of offsite power. The 4 kV emergency bus loss of voltage and degraded voltage relays initiate starting and loading of the emergency diesel generators (EDGs) when the preferred power source voltage is lost or drops below a predetermined value. The relays also initiate disconnection of the preferred power source from the 4 kV emergency busses. These actions ensure adequate terminal voltage to all safety-related electrical equipment required to support accident mitigation. The required voltage necessary to ensure safety-related motors are capable of starting is 75 percent of nominal rated equipment voltage. The required voltage necessary to ensure these motors continue running for extended periods is 90 percent of nominal rated equipment voltage.

The degraded (transient) voltage setpoint is being changed from 3628 [plus or minus] 25 Volts to 3710 [plus or minus] 80 Volts. Based on the most recent calculations, a minimum voltage of 3630 Volts is required to ensure at least 75 percent of the nominal voltage is available to No. 13 Charging Pump, which is the most limiting electrical load.

The new steady state degraded voltage relay setpoint will be established at 3900 [plus or minus] 80 Volts. The setpoint ensures that there is at least 90 percent of nominal voltage available to No. 13 Charging Pump. The time delay associated with this actuation is 101 [plus or minus] 3.5 seconds. The time delay provides adequate time for the voltage regulator to recover bus voltage following a voltage swing on the 500 kV system and time for the EDG voltage regulator to stabilize. The steady state degraded voltage relays will be tested in the same manner, and at the same frequency, as the loss of voltage and transient degraded voltage relays.

The required voltage range during EDG surveillance testing is being revised from 4160 [plus or minus] 420 Volts to 4160 +240, -100 Volts. The surveillance requirement verifies that the EDG voltage regulator is maintaining an acceptable voltage. The

new value ensures the 4 kV motors are operated within their rated voltage and prevents actuation of the steady state degraded voltage relay during surveillance testing.

The degraded voltage relays are not initiators in any previously evaluated accidents. Additionally, decreasing the acceptable voltage range during EDG testing does not affect the initiation of any previously analyzed accidents. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously analyzed.

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

The license amendment request revises the current degraded voltage setpoint and adds an additional steady state degraded voltage requirement. Additionally, the acceptable voltage range during EDG surveillance testing is being decreased. The proposed changes ensure adequate starting and running terminal voltage to all safety-related electrical equipment during steady state and transient degraded voltage conditions. The addition of the steady state degraded voltage relays provide an extra scheme of protection against sustained degraded voltage conditions. The facility currently relies upon degraded voltage relays to start and load the EDGs and to disconnect the preferred power source from the 4 kV emergency busses. Therefore, revising the relay setpoint, adding additional steady state degraded voltage protection, and decreasing the acceptable voltage range during EDG testing does not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The safety function of the degraded voltage relays is to ensure that the preferred power source is disconnected from the 4 kV emergency busses during loss of voltage or degraded voltage conditions. The relays also ensure the EDGs are started and loaded. Ultimately, these actions ensure the minimal terminal voltage necessary to start and run all safety-related electrical equipment is maintained. The proposed changes revise the current degraded voltage setpoint and adds an additional steady state undervoltage requirement. Additionally, the acceptable voltage range during EDG surveillance testing is being decreased to ensure actuation of the steady state degraded voltage relays does not occur during

EDG testing, and to ensure the 4 kV motors are operated within their rated voltage range.

Because the proposed changes ultimately ensure adequate terminal voltage to all safety-related electrical equipment during transient and steady state undervoltage conditions, the safety function of the degraded voltage relays, as well as the margin of safety afforded by these relays is unchanged.

Therefore, the 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

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

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

NRC Project Director: S. Singh Bajwa, Director

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: October 2, 1997

Description of amendment request: The proposed amendment would address an unreviewed safety question associated with the analysis of a fuel handling accident in the Fuel Storage Building as described in Section 15.7.4, ''Design Basis Fuel Handling Accidents,'' of the H.B. Robinson Steam Electric Plant (HBR) Updated Final Safety Analysis Report (UFSAR). Carolina Power & Light Company (the licensee) determined that an assumption used in the accident analysis for depth of water above the top of irradiated fuel in the spent fuel pit was non-conservative. The accident analysis assumed a depth of 23 feet instead of the correct value of 21 feet. The licensee has submitted a revised accident analysis using the correct assumption and has proposed that the UFSAR be changed to incorporate the results of the revised analysis.

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

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

The proposed change to the UFSAR is to change assumptions associated with the evaluation of a fuel handling accident in the Fuel Storage Building. The change in assumptions is to reduce the decontamination factor associated with the removal of elemental iodine from the spent fuel pool water. Because the decontamination factor for elemental iodine is reduced, the consequences of a fuel handling accident in the Fuel Storage Building is [sic] increased. However, because the radiological consequences remain well within the exposure guideline values of 10 CFR 100, paragraph 11 (i.e., 25% or less of the values), the increase in consequences is not significant. The change in assumptions for the fuel handling accident in the Fuel Storage Building do [sic] not affect operation, maintenance, or design of equipment associated with the handling of fuel in the Fuel Storage Building, therefore, the probability of a fuel handling accident as previously evaluated is not changed.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change to the UFSAR to change the assumptions associated with a fuel handling accident in the Fuel Storage Building is to change the assumption for the decontamination factor for elemental iodine to a smaller value. The new assumption for elemental iodine decontamination factor preserves the approximate factor of 24 margin between experimental data for elemental iodine decontamination factor and the assumed value provided in NRC Safety Guide 25. Therefore, the change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

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

NRC Project Director: James E. LyonsCommonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois Date of application for amendment request: September 30, 1997

Description of amendment request: This request changes the Technical Specifications (TS) by adding a new Section 3/4.12.C, ''Inservice Leak and Hydrostatic Testing Operation,'' to allow certain reactor coolant pressure tests to be performed in MODE 4 when the metallurgical characteristics of the reactor pressure vessel require the pressure testing at or approaching temperatures ≤212°F, which normally correspond with MODE 3.

Basis for proposed no significant hazards determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: 1) Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

The proposed amendment represents the addition of a Special Test Exception to perform Pressure Testing Operations consistent with the requirements of Section 3.10.1 of the Improved Standard Technical Specifications (NUREG-1433). The proposed changes are consistent with the current plant safety analyses. Implementation of these changes will provide continued assurance that specified parameters associated with Pressure Testing Operations will remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

The proposed changes are based on requirements specified by Section 3.10.1 of NUREG-1433. Any such changes are consistent with the current plant safety analyses and have been determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analyses, or provide continued assurance that specified parameters associated with Pressure Testing Operations remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems affecting Pressure Testing Operations related to this proposed amendment are not assumed in any analyses to initiate any accident sequence; therefore, the probability of any accident previously evaluated is not increased by this proposed amendment which incorporates the requirements of Section 3.10.1 of NUREG-1433. In addition, the proposed limiting conditions for operation and surveillance requirements for the proposed amendment ensure a level of equipment operability sufficient to mitigate any operational occurrences which could occur while operating under this Special Test Exception. Furthermore, any operational occurrence postulated during operation under this Special Test Exception is bounded by the Design Basis Accidents. Therefore, the proposed amendment does not increase the consequences of nay accident previously evaluated.

There is no change to the consequences of an accident previously evaluated because Pressure Testing Operations does not adversely affect either the on-site or off-site does consequences resulting from an accident. In addition, Pressure Testing Operations is not an accident initiator. As such, there is no adverse impact on the probability of accident initiators. Thus, there is no significant increase in the probability of any previously analyzed accident.

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

The proposed amendment represents the conversion of current Technical Specification requirements to maintain consistency with those requirements specified in Section 3.10.1 of NUREG-1433. The proposed changes are consistent with the current plant safety analyses. These proposed changes do not involve revisions to the design of the station. In addition, the proposed limiting conditions for operation and surveillance requirements for the proposed amendment ensure a level of equipment operability sufficient to mitigate any operational occurrences which could occur while operating under the Special Test Exception. Some of the changes may involve revision in the testing of components at the station; however, these are in accordance with the current plant safety analyses. The proposed changes will not introduce new failure mechanisms beyond those already considered In the current plant safety analyses.

The associated systems that affect Pressure Testing Operations related to the proposed amendment, are not assumed in any plant safety analysis to initiate any accident sequence. In addition, the proposed surveillance requirements for any such affected systems are consistent with the requirements of Section 3.10.1 of NUREG-1433. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

ComEd proposes to revise the Technical Specifications to be consistent with those provisions specified in Section 3.10.1 of NUREG-1433. The proposed changes are consistent with the current plant safety analyses. In addition, these proposed changes do not involve revisions to the design of the station. As such, the proposed individual changes will maintain the same level of reliability of the equipment associated with Pressure Testing Operations, assumed to operate in the plant safety analysis, or provide continued assurance that specified parameters affecting, will remain within their acceptance limits. Therefore, the proposed changes provide continued assurance of Pressure Testing Operations without adversely affecting the public health and safety and as such, will not significantly reduce existing plant safety margins.

The proposed amendment to the Technical Specifications implements present requirements, or the requirements in accordance with the guidelines set forth in Section 3.10.1 of NUREG-1433. The proposed changes have been evaluated and found to be acceptable for use at the stations based on system design, safety analysis requirements, and operational performance. Since the proposed changes are based on NRC accepted provisions that are applicable at the stations and maintain necessary levels of system or component reliability affecting Pressure Testing Operations, 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 are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: September 26, 1997

Description of amendment request: The proposed amendments would revise the Technical Specifications to (1) prohibit the simultaneous opening of the drywell and suppression chamber purge system isolation valves, (2) upgrade the ventilation filter testing program to the latest industry standards, and (3) specify that the auxiliary electric equipment room is required to be habitable during design bases accidents.

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

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

a. Drywell and Suppression Chamber Purge System

The purpose of the drywell and suppression chamber purge system isolation valves is to mitigate the consequences of a design bases accident. Operation of these valves will have no effect on the probability of a design bases accident occurring.

The current TS 3.6.1.8 allows for the drywell and suppression chamber purge system isolation valves to be open simultaneously. In this condition, containment pressure and offsite dose during design bases accidents would be greater than previously evaluated. The proposed revision to TS 3.6.1.8 would prevent the simultaneous opening of the drywell and suppression chamber purge system isolation valves thus assuring that the consequences of design bases accidents previously evaluated are still bounding.

b. Ventilation Filter Testing Program

The SBGTS [Standby Gas Treatment System] and Control Room and AEER [Auxiliary Electric Equipment Room] Emergency Filtration Systems are designed to mitigate the radiological consequences of previously evaluated design bases accidents. Operation and testing of these systems will have no effect on the probability of a design bases accident occurring.

The proposed revisions associated with this change relocate the requirements for SBGTS and Control Room and AEER Emergency Filtration System filter testing from the current TS SRs to a new TS administrative control program. The testing requirements are being upgraded to the latest industry standards. Filter testing in accordance with the proposed program will ensure that Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, General Design Criteria (GDC) 19 and 10 CFR 100 limits are not exceeded.

c. Other Control Room and Auxiliary Electric Equipment Room Emergency Filtration System Changes

The SBGTS and Control Room and AEER Emergency Filtration System are designed to mitigate the radiological consequences of previously evaluated design bases accidents. Operation and testing of these systems will have no effect on the probability of a design bases accident occurring.

The proposed revisions associated with this change acknowledge that the AEERs are required to be habitable during design bases accidents. This is consistent with the plant—s design bases.

d. Editorial Changes

The proposed revisions to TS 6.2.F.7 reformat the requirement to establish consistency with the remainder of TS 6.2.F. There are no technical changes being proposed.

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

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

a. Drywell and Suppression Chamber Purge System

No new plant equipment is being installed, and use of currently installed plant equipment is not affected by this proposed change. The proposed revision to TS 3.6.1.8 provides additional limitations on the opening of the drywell and suppression chamber purge system isolation valves.

b. Ventilation Filter Testing Program

No new plant equipment is being installed, and use of currently installed plant equipment is not affected by this proposed change. These proposed revisions will demonstrate operability of the Control Room and AEER Emergency Filtration System using the latest industry standards.

c. Other Control Room and Auxiliary Electric Equipment Room Emergency Filtration System Changes

No new plant equipment is being installed, and use of currently installed plant equipment is not affected by this proposed change. These proposed revisions will demonstrate habitability of the AEER by imposing operability requirements on the AEER recirculation filter units.

d. Editorial Changes

The proposed revisions to TS 6.2.F.7 reformat the requirement to establish consistency with the remainder of TS 6.2.F. There are no technical changes being proposed.

Based upon the above, the proposed change will not create the possibility of a new or different kind of accident or transient previously evaluated.

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

a. Drywell and Suppression Chamber Purge System

The current TS 3.6.1.8 requirements are non-conservative with respect to the assumptions used when evaluating steam bypass of the suppression chamber; specifically, a maximum allowable leakage area of 0.03 square feet with the only credible leakage path was assumed to be suppression chamber vacuum breaker valve seat leakage. This proposed revision to TS 3.6.1.8 will make the TS requirements consistent with those assumptions.

b. Ventilation Filter Testing Program

These proposed revisions will ensure operability of the Control Room and Auxiliary Electric Equipment Room (AEER) Emergency Filtration system using the latest industry standards. Filter testing in accordance with the proposed program will ensure that GDC 19 and 10 CFR 100 limits are not exceeded.

c. Other Control Room and Auxiliary Electric Equipment Room Emergency Filtration System Changes

These proposed revisions will ensure operably of the control room and AEER Emergency Filtration System by demonstrating system performance with the control room and AEER recirculation filter units to ensure GDC 19 limits are not exceeded.

d. Editorial Changes

The proposed revisions to TS 6.2.F.7 reformat the requirement to establish consistency with the remainder of TS 6.2.F. There are no technical changes being proposed.

Based on the above, the proposed TS change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra

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

Date of amendment request: October 15, 1997

Description of amendment request: The proposed amendments would revise Technical Specification Table 4.3.7.5-1, Accident Monitoring Instrumentation Surveillance Requirements, by deleting a footnote that provides details concerning the calibration requirements for the drywell hydrogen concentration analyzer and monitor.

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

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

The drywell hydrogen concentration analyzer and monitors are required to be operable by TS 3/4.7.5, Accident Monitoring Instrumentation. Table 4.3.7.5-1, Accident Monitoring Instrumentation Surveillance Requirements, includes a footnote providing unnecessary details related to the calibration of this specific analyzer and monitors. The footnote provides information that was determined to put the hydrogen analyzers and monitors outside of the design basis by limiting the range of the indication to 0% to 4% hydrogen in the drywell. The calibration method is being corrected to provide the correct range of 0% to 10%, and requires this note in the TS to be changed or deleted. The footnote is proposed to be deleted from the TS, because it provides unnecessary detail.

Deletion of the footnote will not cause an increase in the probability of an accident, because this instrumentation is only for accident monitoring instrumentation and thus does not affect accident initiators or assumptions.

Deletion of the footnote will not change the consequences of an accident previously evaluated, because this detail in the TS does not change the requirement of performing a channel calibration at the specified frequency. In addition, the ability to monitor hydrogen during an accident will not be affected by deletion of the footnote.

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

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

This is monitoring instrumentation only. Deletion of the footnote concerning specifics on how to calibrate this instrumentation will not affect the reliability or failure modes of the drywell hydrogen concentration analyzer and monitors. Therefore, this 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 the margin of safety because:

This is monitoring instrumentation only. Deletion of the footnote concerning specifics on how to calibrate this instrumentation will not change the requirement to perform Channel Calibrations at the frequency specified in the TS. The details of how to perform a Channel Calibration on the drywell hydrogen concentration analyzer and monitors are located in plant procedures and are in accordance with vendor recommendations. The TS requirements for redundancy of the instrumentation and the actions to be taken for inoperable instrumentation are also not affected by the deletion of this footnote.

This change to the level of information regarding this calibration is consistent with the detail for this and other instrumentation in NUREG-1434, Revision 1, Standard Technical Specifications, General Electric Plants, BWR/6.

Therefore, deletion of footnote * from TS Table 4.3.7.5-1 will not involve a reduction in the margin of safety.

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

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

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

NRC Project Director: Robert A. Capra

Duke Energy Corporation, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: October 6, 1997

Description of amendment request: The proposed amendments would delete all references to the steam line low pressure safety injection function.

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

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

Answer

Probability

Accident initiators can affect the probability of a previously evaluated accident. The addition of a new device or piece of equipment to the plant may introduce a new accident initiator. No new equipment is added to the plant as a result of this change. The proposed removal of the low steam line steam pressure will involve removing the steam line pressure safety injection function. This results in a reduction in the likelihood of spurious safety injections. Spurious safety injections can result in inadvertent ECCS [emergency core cooling system] actuations. Inadvertent ECCS Actuation is a UFSAR [updated Final Safety Analysis Report] accident (UFSAR 15.5.1). Therefore, this change will result in a reduction in the probability of an accident previously evaluated.

Routine plant operating practices and conditions will not be altered by the removal of the safety injection function. Therefore, there is no operating practice or condition change that could increase the probability of occurrence of a previously evaluated accident.

There is no significant increase in the probability of an accident previously evaluated.

Consequences

Accidents previously evaluated that could be adversely affected are the steam line break and the feedwater line break. These accidents will result in secondary side depressurization with pressure reaching the current actuation setpoint. The review of these accidents found that the consequences of the previous accident analysis acceptance criteria remain satisfied. The specifics of the accident analysis is discussed below.

The steam line break accident was analyzed to demonstrate short term cooling capability. A spectrum of break sizes were evaluated to determine the limiting break size. For smaller breaks (including the limiting break size), the safety injection actuation on low pressurizer pressure occurs prior to low steam line pressure safety injection. However, for larger steam line breaks the setpoint for low steam line pressure safety injection is reached prior to low pressurizer pressure safety injection. The larger spectrum of breaks were analyzed without credit for the low steam line pressure safety injection. The results of this analysis found that there would be a slight increase in time required for safety injection to actuate. The low pressurizer safety injection would actuate in these accidents due to the cooldown and depressurization of the reactor coolant system in response to the secondary side energy removal. The Departure from

Nucleate Boiling Ratios (DNBRs) were analyzed with this time delay in safety injection. The DNBRs for these cases were found to be less limiting than those calculated for the limiting break size. Therefore, the removal of steam line low pressure safety injection does not adversely affect the DNBR, fuel failure or dose consequences of the main steam line break accident. Other acceptance criteria would not be expected to be affected by the small change in timing of the safety injection signal.

In addition, to the Chapter 15 accident analysis, the Chapter 6 containment response to mass and energy releases was evaluated without credit for steam line low pressure safety injection. The evaluation demonstrated that for steam line breaks inside of containment, the high containment pressure safety injection set point is reached prior to the pressure associated with steam line low pressure safety injection. Therefore the existing containment response evaluation is not adversely affected by the removal of the low steam pressure safety injection. This also assures that the existing environmental qualification envelope for McGuire is not affected by this change. For steam line breaks outside of containment the maximum required breaksize is 1.0 ft2, which results in transients with safety injection caused by low pressurizer pressure prior to low steam line pressure safety injection.

The feedwater line break accidents were analyzed to demonstrate long term core cooling capability. During a feedwater line break, the secondary system will depressurize if the break occurs between the main feedwater check valve and the steam generator. However, breaks are only required to occur at the terminal ends of feedwater piping (i.e., at the feedwater pump or at the steam generator). For a feedwater line break at the main feedwater pump, the main feed check valve will prevent depressurization of the steam generator. For a feedwater line break at the steam generator, a safety injection on high containment pressure will occur prior to safety injection on steam pressure. Therefore, the elimination of the steam line low pressure safety injection does not adversely impact the feedwater line break accident.

In summary, a review was conducted of all design basis accidents to identify those which result in a low steam pressure safety injection. These accidents were then evaluated to verify that the accident analysis were within acceptance criteria. This review revealed that all accident analysis results were within current analysis acceptance criteria.

Therefore, there is no significant increase in the consequences of a previously evaluated accident.

Conclusion

Elimination of the low steam line pressure safety injection results in no 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[?]

Answer

There is no introduction of new equipment or operating practices that could result in a new operating condition. The plant will continue to operate in the same method with the same complement of equipment with the exception of the actuation logic associated with the steam line low pressure safety injection. Therefore, there is no new operating condition that would be expected to generate a new sequence of events which could generate a new or different accident. There is no new equipment that could interact with other plant structures, systems or components.

The low pressure safety injection equipment is the only plant equipment affected by this change. There are no new equipment failure modes which might result in a new or different accident.

Affected accidents were evaluated to validate that the accident sequence would not deviate in a fashion which would create a new or different accident. The analysis of the feedwater line break and steam line break did not reveal any new or different type of accident.

Removal of the low steam line pressure safety injection will not create the possibility of a new or different kind of accident from any accident previously evaluated;

(OR)

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

Answer

The margin of safety relevant to this change is represented by the margin of physical protection provided by fuel cladding and the reactor containment. Effects of this change on the safety analysis was described under question 1 above. The results of the analysis demonstrate that DNBR, fuel clad integrity and containment response were not significantly affected by the removal of low steam line pressure safety injection. Therefore, the physical protection provide[d] by the fuel cladding and reactor containment were not affected by this change. Accident acceptance criteria continued to be met without credit for the safety function. The radiological consequences of accidents was not affected by the change.

The removal of the low steam line pressure safety injection did not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina

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: October 15, 1997

Description of amendment request: The proposed amendment would affect nominal trip setpoints and allowable values for Reactor Trip System (RTS) Instrumentation Trip Setpoints Table 2.2-1, and Engineered Safety Features Actuation System (ESFAS) Instrumentation Trip Setpoints Table 3.3-4. In addition, the proposed amendment would (1) decrease the reactor trip setpoint for the reactor coolant pump (RCP) low shaft speed (underspeed trip setpoint) from 95.8 percent to 92.4 percent of rated speed, (2) make editorial changes, and (3) change the Bases to reflect the new methodology.

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

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

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

The proposed changes to Tables 2.2-1 and 3.3-4 involve changes from a five column format to a two column format. The RTS trip setpoints and ESFAS trip setpoints remain unchanged with the exception of the RCP low shaft speed trip setpoint discussed below. Detailed operability criteria will be moved to surveillance procedures and analysis has demonstrated that an adequate margin for normal trip setpoints exist and safety analysis limits are preserved in all RTS/ESFAS functions.

Changing the RCP low shaft speed trip setpoint will not change the probability of occurrence of the event. The existing accident analysis (Millstone Unit No. 3 FSAR [final safety analysis report] section 15.3.2) of the complete loss of forced reactor coolant flow remains valid for the proposed change. Therefore, the change to the RCP low shaft speed trip setpoint does not increase the probability or consequences of any previously analyzed accident.

In addition, the proposed changes to Tables 2.2-1 and 3.3-4 do not alter the intent or method by which the surveillances are conducted. Therefore, the scope of evaluation performed gives reasonable assurance that there will not be an adverse impact on the consequences or the probability of any previously analyzed accident.

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 existing design basis adequately covers the plant response with the proposed change to the RCP low shaft speed trip setpoint. The change does not introduce new failure modes.

The proposed changes to Tables 2.2-1 and 3.3-4 do not modify the design or operation of any plant system. The proposed changes do not alter the intent or method by which the surveillances are conducted, other than adjusting the allowable values to reflect historical instrument performance data. 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 proposed changes to Tables 2.2-1 and 3.3-4 modify the existing five column format to a two column format to show the RTS and ESFAS nominal trip setpoints and the process rack bistable allowable values for individual functions. Detailed operability criteria will be moved to the surveillance procedures. With the exception of the low shaft speed trip discussed below, the RTS and ESFAS setpoints remain unchanged and analysis has demonstrated that an adequate margin for normal trip setpoints exist and safety analysis limits are preserved in all RTS/ESFAS functions.

Since the safety limits of the design are still met, the proposed change to the RCP low shaft speed trip setpoint does not reduce the margin of safety.

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, Connecticut

NRC Deputy Director: Phillip F. McKee

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

Date of amendment requests: March 10, 1997, as supplemented by letter dated May 20, 1997

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant Unit Nos. 1 and 2 to revise TS 3/4.4.5 and 3.4.6.2, including associated Bases 3/4.4.5 and 3/4.4.6.2, to allow the implementation of steam generator (SG) tube alternate repair criteria for axial indications in the Westinghouse explosive tube expansion (WEXTEX) region below the top of the tubesheet and below the bottom of the WEXTEX transition that may exceed the current TS depth-based plugging limit. The allowed primary-to-secondary operational leakage from any one SG would be reduced from 500 gpd to 150 gpd.

Basis for proposed no significant hazards 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.

Probability

Of the various accidents previously evaluated, the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Series 51 steam generators has shown that axial loading of the tubes is negligible during an SSE.

For the SGTR event, the required structural margins of the steam generator tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the Westinghouse explosive tube expansion (WEXTEX) region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' margins against burst are maintained for both normal and postulated accident conditions.

The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident. 1

The proposed changes do not affect other systems, structures, components or operational features. Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

Consequences

The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) in the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage, that is, the combined leakage for all such tubes, plus the combined leakage developed by any other ARC, must be below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR 100 guideline values.

Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the proposed steam generator alternate tube plugging criteria.

WCAP-14797, Revision 1, ''Generic W* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEX Expansions,'' requires that any tubes with indications identified using the bobbin coil probe during the bobbin sampling plan also be inspected with the RPC coil throughout the W* length of the tubes. The use of the RPC will: (a) identify any new or non-expected degradation mode that may not be identified using the bobbin coil probe, and (b) confirm and characterize the bobbin coil indication.

These changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions. RG 1.121 is used as the basis in the development of the W* alternate tube plugging criteria for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of an SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code.

For primarily axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP-14797 defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Application of the W* criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the W* criteria.

Plugging of the steam generator tubes reduces the reactor coolant flow margin for core cooling. Implementation of the proposed changes are expected to result in plugging of fewer tubes than with the current criteria. Thus, implementation of the proposed changes will maintain the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the FSAR Update or bases of the plant Technical Specifications.

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

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

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

NRC Project Director: William H. Bateman

Philadelphia Electric Company, Docket No. 50-352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of amendment request: October 24, 1997

Description of amendment request: The proposed Technical Specifications (TS) changes would revise TS Section 3/4.1.3.6 to exempt control rod 50-27 from the coupling test for the remainder of Cycle 7.

Basis for proposed no significant hazards 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 Technical Specifications (TS) change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of occurrence of the analyzed Control Rod Drop Accident (CRDA) is not increased by operating with the subject control blade in a condition not known to be coupled since the compensatory measures will assure that the blade will remain fully inserted below 10% rated thermal power where the CRDA is a concern. Monitoring of nuclear instrumentation responses in the vicinity of the blade when the drive is withdrawn above 10% power will assure the blade is tracking with the drive with no potential to stick and then drop. Scram impact forces from an uncoupled control rod are of insufficient energy to dislodge the fuel support (or fuel) or to cause a threat to the pressure boundary integrity. No reduction of system or equipment redundancy is involved.

The CRDA analyzed in the Safety Analysis Report (SAR) remains the limiting rod drop accident, and its consequences are unaffected by operation of the subject blade in the proposed manner. Operation of the control blade as described, i.e., withdrawn no further than the 46 position and in a condition not known to be coupled, has no adverse effect on scram performance in response to any other postulated accident. The scram insert motion of the rod is not affected by the potentially uncoupled condition, and since the rod is already partially inserted at position 46, it should have a slightly better negative reactivity insertion characteristic. Therefore, no potential to increase onsite or offsite radiological consequences beyond those previously analyzed in the SAR is created.

Operating the subject control blade in a condition not known to be coupled does not result in any onsite or offsite radiological consequences different from those previously analyzed in the SAR. The subject control blade will be fully inserted below 10% thermal power where the CRDA is a concern and will be monitored during drive withdrawal above 10% thermal power to assure it is tracking with the drive. Scram performance is not adversely affected by operation from the near full-out position of 46. Hence, no new failure modes are created and consequences of any postulated failures are not increased.

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

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

The Safety Analysis Report (SAR) analyzed Control Rod Drop Accident (CRDA) remains the only type of accident initiated (or contributed to) by the control rod drive/control blade interface. The compensatory actions to be taken when operating the subject blade in a condition not verified to be coupled assure that no new types of accidents can occur. The subject control blade will be fully inserted below 10% thermal power where the CRDA is a concern and will be monitored during drive withdrawal above 10% thermal power to assure it is tracking with the drive. Scram performance is not adversely affected by operation from the near full-out position of 46. Since no adverse effect on insertion or scram performance is expected, the previously analyzed accidents encompass any potential consequence of operating with an uncoupled control blade.

The compensatory actions to be taken when operating the subject blade in a condition not verified to be coupled assure that no new failure modes are created, and, therefore, no new type of equipment malfunction is introduced by operating the subject control blade in the proposed manner.

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

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

Operation with the subject control blade in a condition not known to be coupled for the remainder of Cycle 7 at LGS [Limerick Generating Station] Unit 1, but with the compensatory actions described below, does not reduce the existing margin of safety determined by the analysis of the Control Rod Drop Accident (CRDA). The CRDA analyzed in the Safety Analysis Report (SAR) remains bounding in that the subject rod will be fully inserted below 10% rated thermal power where the CRDA is a concern. Above 10% power, when the associated drive is withdrawn, the nuclear instrumentation in the vicinity of the blade will be monitored to assure the blade tracks with the drive, providing assurance that the position of the blade can be ascertained by the drive position. If the control blade can not be verified to have followed the drive, then the rod shall be completely inserted and the control rod directional valves disarmed in accordance with existing TS requirements. To minimize any scram impact loadings, the blade will be operated at the near full-out position of 46 except for intermediate positions temporarily occupied during standard rod withdrawal sequences. Operating the subject control blade in the proposed manner will have no adverse effect on insertion or scram performance of the blade and will preserve the margin of safety.

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

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

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

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, PA 19101

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: October 6, 1997

Description of amendment request: The amendment to the Technical Specifications would increase the allowable band for control and shutdown rod demanded position versus indicated position from plus or minus 12 steps to plus or minus 18 steps when the power level is not greater than 85% rated thermal power. The amendment is identical to Amendment 183 for Salem Unit 2, which was issued September 10, 1997, as an exigent amendment.

Basis for proposed no significant hazards 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 to the rod misalignment criteria of [plus or minus] 18 steps for core powers equal to or below 85% of RATED THERMAL POWER (RTP) does not increase the probability of previously evaluated accidents. Increasing the magnitude of the allowed control rod misalignment is not a contributor to the mechanistic cause of an accident evaluated in any accident analysis. The magnitude of control rod indicated misalignment is a parameter used to establish the initial conditions for accident evaluation.

The proposed increase in the allowable rod misalignment from the current [plus or minus] 12 steps for reactor powers equal to or less than 85% RTP does not involve a significant increase in the consequence of any previously evaluated accident. Rod misalignment affects power distribution, shutdown margin and the ejected rod accident. An extension of the allowable rod misalignment above and below 85% RTP has been analyzed in Westinghouse WCAP-14672. As provided in WCAP-14672, above 85% the allowable misalignment is governed by the available peaking factor margins as determined by flux maps.

[Public Service Electric & Gas] PSE&G is simplifying the proposed change by keeping the currently allowed [plus or minus] 12 step misalignment in Technical Specifications 3.1.3.1 and 3.1.3.2.1 for reactor power greater than 85% RTP.

The PSE&G proposed change is to allow [plus or minus] 18 steps misalignments in Technical Specifications 3.1.3.1 and 3.1.3.2.1 for reactor power less than or equal to 85% RTP. As demonstrated in WCAP-14672, for reactor powers less than 85% RTP, the available peaking factor margin increases faster than any penalty associated with a [plus or minus] 18 step misalignment.

As described in Section 4.0 of the Westinghouse WCAP, a conservative penalty factor has been applied to the rod insertion allowance (RIA) of the shutdown margin calculation to account for rods misaligned an additional [plus or minus] 6 steps (for a total of [plus or minus] 18 steps). This conservative penalty factor is applied as part of the reload analysis in order to satisfy Technical Specification 3.1.1.1.

In addition to the normal, or Condition 1, operational transients, the impacts of increased rod misalignment on Condition II, III and IV accident analysis have also been evaluated. The proposed increase in rod misalignment does not have a significant effect on any moderator or Doppler reactivity coefficients or defects, boron worth or reactor kinetics parameters.

To account for the potential increase in ejected rod parameters, conservative penalty factors have been applied to the reload safety evaluation to cover the additional [plus or minus] 6 step misalignment. Margin is available in the reload safety analysis to accommodate this impact.

Therefore, the proposed amendment does not increase the probability or consequences of any accident previously evaluated.

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

0No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change to the rod misalignment criteria of [plus or minus] 18 steps below 85% RTP. The implementation of the proposed rod misalignment criteria will have no adverse effect on the performance of any other safety related system. Therefore, the proposed amendment 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.

Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety. The Technical Specifications allowed increase in peaking factors as power is reduced accommodates the peaking factor penalty associated with the additional [plus or minus] 6 step misalignment for core powers equal to or less than 85% RTP. Therefore, there is no change to the peaking factors assumed in the safety analysis. In addition to peaking factors, there is no change in any other current limit input into the safety analysis. As the input, or initial conditions, of the safety analysis have not changed, there is no reduction in the margin to 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: Salem Free Public Library, 112 West Broadway, Salem, NJ 08079

Public Service Electric & Gas Company, Docket No. 50-272, Salem Nuclear Generating Station, Unit No. 1, Salem County, New Jersey

Date of amendment request: October 14, 1997

Description of amendment request: The proposed amendment will modify the Salem Unit 1 Technical Specification (TS) 3.4.6.3, ''Primary Coolant System Pressure Isolation Valves Limiting Condition for Operation,'' to be consistent with Salem Unit 2 TSs.

Basis for proposed no significant hazards 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 majority of the proposed changes, as described above, are editorial in nature. Rewording, and reformatting the Limiting Condition for Operation, including the surveillance requirements do not involve a significant increase to the probability or consequences of an accident.

Those substantive changes involving the addition of (1) new reactor coolant system pressure isolation valves, (2) providing for a shorter test frequency upon entry into Mode 4, and (3) adding a new surveillance test requirement, do not increase the probability or consequences of an accident. These changes ensure that the system and components needed to prevent and minimize the effects of inter-system loss of coolant are properly identified in the Technical Specifications.

Although pressure isolation valves are being added to the Technical Specification table, these valves were already included in the IST [inservice testing] program as pressure isolation valves and were being tested as such. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change, as described above, does not physically alter the facility or the operation of the facility. The majority of the changes are editorial in nature and provide for improvement in the human factors of the Technical Specifications, while properly identifying all the pressure isolation valves in the Technical Specifications. The addition of valves into the Technical Specification is an administrative change that improves the quality of the LCO [limiting condition for operation], but does not add components to the facility.

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

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

The margin of safety, as defined in the bases for any technical specifications, depend upon proper identification of equipment and performance of the proper surveillance requirements to demonstrate equipment operability. The proposed change will ensure that the proper valves are identified and tested in accordance with the Technical Specification requirements.

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

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

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

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

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

Date of amendment request: October 21, 1997

Description of amendment request: The proposed amendment revises Technical Specification Tables 3.3-1 and 4.3-1 to require that Functional Unit, ≥2. Power Range, Neutron Flux,'' be operable in Mode 3, as well as in Modes 1 and 2. The change is being proposed because the licensee has determined that the power range nuclear instrumentation should be operable in Mode 3 whenever the reactor trip system breakers are in the closed position and the control rods are capable of being withdrawn.

Basis for proposed no significant hazards 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 requirement for operability of a trip and the surveillance requirements to ensure the functionality of the trip are independent of the probability of an accident previously evaluated. The accident that this trip is intended to mitigate is the Rod Withdrawal from Subcriticality event. The surveillance procedure and the requirement for the trip to be operational when the Control Rod Drive System is capable of rod movement mitigate the consequences of this event, and do not increase the probability of a rod withdrawal from subcritical.

Therefore, the probability and consequences of an accident previously evaluated are not significantly increased.

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

The proposed changes do not involve any modifications to existing plant equipment, do not alter the function of any plant systems, do not introduce any new operating configurations or new modes of plant operation, or change the safety analyses. The proposed change is intended to ensure that the trip function is available and will perform as designed in the event of a previously evaluated event.

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

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

The proposed change does not reduce the margin of safety, because assurance of the operability of the trip function is increased by the proposed change.

Based on the above, PSE&G [Public Service Electric & Gas Company] has determined that the proposed changes do not involve a significant hazards consideration.

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

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, NJ

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

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

Date of amendment request: August 28, 1997

Description of amendment request: The proposed change would revise Technical Specification 3.4.11, ''Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits,'' to incorporate the new P/T curves, which were provided by General Electric Nuclear Energy in report number GE-NE-B1301793-01, ''Perry Unit 1 RPV Surveillance Materials Testing and Analysis.''

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

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

The proposed change will provide for approved P/T limit curves which are valid through 9 effective full-power years (EFPY) and 18 EFPY. This change will not affect any Safety Limits, Power Distribution Limits, or Limiting Conditions for Operation. The proposed changes incorporate operating limits which provide margin to brittle failure of the reactor vessel based on testing of the irradiated reactor vessel materials (base metal, weld material, and heat affected zone material). The limits ensure that adequate safety margins against nonductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The specimens have been tested and analyzed in accordance with 10 CFR 50, Appendices G and H, using the methods described in Generic Letter 88-11 and Regulatory Guide 1.99 Revision 2. The predicted lowest upper shelf energy at 32 EFPY was greater than the minimum required by 10 CFR 50, Appendix G. The adjusted reference temperature for the limiting material was lower than the 200 degree Fahrenheit limit required by Regulatory Guide 1.99 Revision 2. As such, the integrity of the reactor pressure coolant boundary is maintained. The changes will result in equivalent or more conservative limits on reactor vessel pressure as a function of temperature for all operational conditions (hydrostatic and leak testing, non-nuclear heatup/cooldown, and core critical operations). The methodology used to derive these values produces limits which continue to ensure that sufficient margin is maintained to meet the criteria of GDC 31, ''Fracture Prevention of Reactor Coolant Pressure Boundary.'' There are no plant modifications associated with this change and no new or revised system interfaces. The proposed changes do not increase the probability of occurrence or consequences previously evaluated because the temperature shifts are well within equipment operating ranges. As such, there is no increase in the probability of occurrence or the consequences of an accident previously evaluated.

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not involve any new modes of operation. The only change will be operation of the plant within operating pressure limits which are determined in a more conservative manner. Therefore, no new failure mode or accident sequence is introduced by this change.

The testing and analysis meets 10 CFR 50, Appendices G and H, requirements; therefore, no new accident types, such as brittle fracture of a reactor pressure coolant boundary component is postulated. The adjusted reference temperature and upper shelf energy predicted at 32 EFPY are well within the limits of 10 CFR 50, Appendices G and H. Therefore, the possibility of an accident of a new or different type than any previously evaluated is not created.

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

The P/T limits are established to provide acceptable margins for the operation of the reactor coolant system during heat up and cool down, criticality, and hydrotest conditions. Technical Specification 3.4.11 limits the rates of change of temperature and pressure to values consistent with the fracture toughness requirements of 10 CFR 50, Appendices G and H, and ASME Boiler and Pressure Vessel Code Section III Appendix G. The bases section for Technical Specification 3.4.11 refers to 10 CFR 50, Appendices G and H, and ASME Code Section III Appendix G. Changes in these limits are necessary because the fracture toughness properties of ferritic materials in the reactor vessel change as a function of reactor operating time. The specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in developing the P/T limits are defined by 10 CFR 50, Appendices G and H. The specific limits defined by 10 CFR 50, Appendices G and H, set the margin of safety for the reactor pressure vessel coolant boundary. Since the testing and analysis of the vessel specimens meet the requirements and limits defined in 10 CFR 50, Appendices G and H, the margin of safety as defined in the basis for Technical Specification 3.4.11 is not reduced. The revised curves are based on the latest NRC guidelines along with actual neutron fluence data for Perry. The new limits conservatively account for irradiation embrittlement effects, thereby maintaining the margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: September 8, 1997

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.2.2.e, ''Organization - Unit Staff,'' by removing the reference to the NRC Policy Statement on working hours. Administrative procedures will be developed to limit the working hours of unit staff who perform safety-related functions.

Basis for proposed no significant hazards 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 to TS 5.2.2.e only alters the administrative location of and the regulatory controls applicable to unit staff specific overtime limits and working hours. Overtime will remain controlled by plant administrative procedures. Changes to the relocated overtime limits and working hours will be subject to review and evaluation under 10 CFR 50.59, ''Changes, Tests and Experiments.'' There is not an increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

There is not an increase in the radiological consequences of an accident previously evaluated because the proposed change does not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed change does not alter the source term, containment isolation, or allowable radiological releases. Therefore, there is no increase in the radiological consequences of an accident previously evaluated.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not change the way the plant is operated, and no new or different failure modes have been defined for any plant system or component important to safety, nor has any limiting single failure been identified as a result of the proposed change. No new or different types of failures or accident initiators are introduced by the proposed change.

The proposed change to TS 5.2.2.e only alters the administrative location of and the regulatory controls applicable to unit staff specific overtime limits and working hours. Therefore, there is no possibility created for a new or different kind of accident.

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

The proposed change does not involve a reduction in a margin of safety because unit staff overtime is not an input in the calculation of a safety margin with regard to Technical Specification Safety Limits, Limiting Safety System Settings, other Technical Specification Limiting Conditions for Operation, the Operational Requirements Manual, or other previously defined margins for any structure, system, or component important to safety. The proposed change to TS 5.2.2.e only alters the administrative location of and the regulatory controls applicable to unit staff specific overtime limits and working hours.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: October 24, 1997

Brief description of amendments: Change to the core safety limit curves and overtemperature N-16 reactor trip function setpoints to support operation with Unit 1, cycle 7 core configuration.

Basis for proposed no significant hazards 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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

A. Revision to the Unit 1 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed the analyses of the Unit 1, Cycle 7 core configuration to determine the reactor core safety limits. The methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a different temperature and power level might be attained than if the plant were operating within the bounds of the old curves.

However, since the new curves were developed using NRC approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification BASES for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

B. Revision to Unit 1 Overtemperature N-16 Reactor Trip Setpoints

As a result of changes discussed, the Overtemperature reactor trip setpoint has been recalculated. These trip setpoints help ensure that the core safety limits are protected and that all applicable limits of the safety analysis are met.

Based on the calculations performed, no significant changes to the safety analysis values for Overtemperature reactor trip setpoint were required. The f(deltaI) trip reset function was revised due to more top-skewed axial power distributions predicted for this cycle. The analyses performed show that, using the TU Electric methodologies, all applicable limits of the safety analysis are met. This setpoint provides a trip function which allows the mitigation of postulated accidents and has no impact on accident initiation. Therefore, the changes in safety analysis values do not involve an increase in the probability of an accident and, based on satisfying all applicable safety analysis limits, there is no significant increase in the consequences of any accident previously evaluated.

In addition, sufficient operating margin has been maintained in the overtemperature setpoint such that the risk of turbine runbacks or reactor trips due to upper plenum flow anomalies or other operational transients will be minimized, thereby, reducing potential challenges to the plant safety systems.

SUMMARY

The changes in the amendment request applies NRC approved methodologies to changes in safety analysis values, new core safety limits and new N-16 setpoint and parameter values to assure that all applicable safety analysis limits have been met. The potential for an operational transient to occur has not been affected and there has been no significant impact on the consequences of any accident previously evaluated.

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

The proposed changes involve the calculation of new reactor core safety limits and overtemperature reactor trip setpoint resets. As such, the changes play an important role in the analysis of postulated accidents but none of the changes effect plant hardware or the operation of plant systems in a way that could initiate an accident. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

In reviewing and approving the methods used for safety analyses and calculations, the NRC has approved the safety analysis limits which establish the margin of safety to be maintained. While the actual impact on safety is discussed in response to question 1, the impact on margin of safety is discussed below:

A. Revision to the Unit 1 Reactor Core Safety Limits

The NRC-approved TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 1, Cycle 7, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The acceptance criteria remains valid and continues to be satisfied; therefore, no change in a margin of safety occurs.

B. Revision to Unit 1 Overtemperature N-16 Reactor Trip Setpoints

Because the reactor core safety limits for CPSES Unit 1, Cycle 7 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the reactor core safety limits must also be recalculated. The Overtemperature N-16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the Overtemperature reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has been analyzed with the new safety analysis value for the Overtemperature reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint (Technical Specification Table 2.2-1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2-1. Meeting the requirements of Technical Specification Table 2.2-1 assures that the Overtemperature reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 1, Cycle 7 Overtemperature reactor trip setpoint is not significantly different from the previous cycle, and thus provides operational flexibility to withstand mild transients without initiating automatic protective actions. Although the value of the f(deltaI) trip reset function setpoint is different, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint are consistent with the safety analysis assumptions which have been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

Using the NRC approved TU Electric methods, the reactor core safety limits are determined such that all applicable limits of the safety analyses are met. Because the applicable event acceptance criteria continue to be met, there is no significant reduction in the margin of safety.

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

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

Attorney for licensee: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, N.W., Washington, DC 20036

NRC Project Director: James W. Clifford, Acting

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

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

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

Duke Energy Corporation, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: October 13, 1997

Description of amendment request: The proposed amendments would support replacement of the three safety-related wide range level instruments. The engineered safety features trip setpoint for the refueling water automatic switchover to recirculation would be revised to account for the difference in instrument uncertainty associated with wide range level instruments and provide additional response time margin.

Date of publication of individual notice in **Federal Register:** October 22, 1997 (62 FR 54859)

Expiration date of individual notice: November 21, 1997

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina

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 amendment request: October 20, 1997

Description of amendment request: The proposed amendments would allow use of a rerolling process as an additional repair method for tube degradation found in the tubesheet region. The rerolling method is designed to ensure that the area of degradation will not serve as a pressure boundary once the repair roll is installed, thus permitting the tube to remain in service.

Date of publication of individual notice in **Federal Register:** October 28, 1997 (62 FR 55835)

Expiration date of individual notice: Comment period ends November 12, 1997; Notice period ends November 28, 1997

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

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.

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: October 4, 1997

Brief description of amendments: These amendments revise the surveillance requirements in Technical Specifications (TSs) 4.1.2.3.1, 4.1.2.4.1, 4.5.2.b, and 4.6.2.1.b and associated Bases. The subject surveillance requirements are applicable to the charging/high-head safety injection pumps, low-head safety injection pumps, and the containment quench spray pumps. The proposed changes replace the current specific test acceptance criteria contained in these surveillance requirements with requirements to verify pump performance in accordance with the inservice testing program, the emergency core cooling system flow analysis, or the containment integrity safety analysis, as applicable. The proposed changes also make minor editorial changes in these TSs and make conforming changes in the TS Index pages.

Date of issuance: October 28, 1997

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

Amendment Nos.: 207, 86

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

Date of initial notice in **Federal Register:** December 18, 1996 (61 FR 66706) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 28, 1997. No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: May 27, 1997, supplemented by October 6, 1997

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

Date of issuance: November 5, 1997

Effective date: November 5, 1997

Amendment No.: 133

Facility Operating License No. NPF-29: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33122) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 1997.No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 17, 1996, as supplemented October 14, 1997

Brief description of amendment: The amendment revises

Facility Operating License No. NPF-38 to reflect the name change from Louisiana Power & Light Company to Entergy Louisiana, Inc.

Date of issuance: November 3, 1997

Effective date: November 3, 1997, to be implemented within 60 days.

Amendment No.: 134

Facility Operating License No. NPF-38: Amendment revised

Facility Operating License No. NPF-38.

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 29749) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 3, 1997. The letter dated October 14, 1997, provided clarifying information which did not alter the initial no significant hazards determination. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 27, as supplemented April 3, May 1, and August 20, 1997.

Brief description of amendment: Change Technical Specifications (TS) to permanently establish a primary-to-secondary leak rate of 150 gallons per day through any one steam generator and specify the steam generator tube inservice inspection requirements for pit-like intergranular attack degradation in the ''B'' Once-Through-Steam-Generator.

Date of issuance: October 28, 1997

Effective date: October 28, 1997

Amendment No.: 158

Facility Operating License No. DPR-72: Amendment revised the TS.

Date of initial notice in **Federal Register:** June 4, 1997 (62 FR 30632) The August 20, 1997, letter provided clarifying information that did not affect the initial no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

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

Date of application for amendments: August 27, 1997

Brief description of amendments: The admendments change the Administrative Section of the Technical Specifications (TS) to allow the use of 12-hour shifts.

Date of issuance: October 27, 1997

Effective date: October 27, 1997

Amendment Nos: 194 and 188Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the TS.

Date of initial notice in **Federal Register:** September 24, 1997 (62 FR 50006) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 27, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 26, 1997.

Brief description of amendments: The amendments modify surveillance 4.7.5.1.e.2 which requires verification of the control room ventilation system autostart function.

Date of issuance: October 28, 1997

Effective date: October 28, 1997, with full implementation within 45 days.

Amendment Nos.: 218 and 202

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

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27796) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 28, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 15, 1997

Brief description of amendment: The amendment revises Technical Specification Sections 3.1 and 4.1, ''Reactor Protection System,'' and the associated Bases to remove run mode intermediate range monitor high flux/inoperative with the associated average power range monitor downscale scram trip function. The amendment also makes other editorial revisions.

Date of issuance: October 27, 1997

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

Amendment No.: 103

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33127) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: September 2, 1997

Brief description of amendment: The amendment changes the Technical Specifications (TSs) by modifying the maximum allowed primary containment internal pressure during normal operation from 2.1 pounds per square inch gauge (psig) to 1.0 psig. The TS Bases, Section 3/4.6.1.4, is also updated to reflect the new maximum allowed primary containment internal pressure during normal operation.

Date of issuance: October 27, 1997

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

Amendment No.: 209

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

Date of initial notice in **Federal Register:** September 24, 1997 (62 FR 50007) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 27, 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, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut

Northern States Power Company, Docket No. 50-263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: November 25, 1996, as supplemented December 12, 1996, April 23, May 8, July 1, August 21, and September 29, 1997

Brief description of amendment: The amendment modifies the Technical Specification requirements associated with the Minimum Critical Power Ratio (MCPR) safety limits for Cycle 18 based on the cycle-specific analysis of the current mixed core of GE11/GE10 fuel parameters.

Date of issuance: October 29, 1997

Effective date: October 29, 1997

Amendment No.: 99

Facility Operating License No. DPR-22: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17238) The December 12, 1996, letter provided an affidavit for the original application dated November 25, 1996. The April 23, May 8, August 21, and September 29, 1997, letters provided clarifying information in response to the staff's request for additional information during a teleconference on March 18, 1997. The July 1, 1997, letter provided a nonproprietary version of the April 23, 1997, submittal. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards considerations determination. Therefore, renoticing was not warranted. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 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

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: November 27, 1996, as supplemented August 15, September 2, and October 3, 1997

Brief description of amendments: The amendments incorporate Combustion Engineering steam generator tube sleeve designs and installation and examination techniques into the plant Technical Specifications (TS). Specifically, the amendments make changes to TS 4.12, ''Steam Generator Tube Surveillance,'' and its associated Bases Section B.4.12, ''Steam Generator Tube Surveillance.''

Date of issuance: November 4, 1997

Effective date: November 4, 1997, with full implementation within 30 days

Amendment Nos.: 132 and 124

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43370) The August 15, September 2, and October 3, 1997, letters provided clarifying information and updated TS pages. This information was within the scope of the original application and did not change the staff's initial no significant hazards considerations determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 4, 1997.

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

PECO Energy Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: June 4, 1997

Brief description of amendments: The proposed change revises the Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, Technical Specifications to eliminate an inconsistency between emergency core cooling system (ECCS) operability requirements and the auto-start and protective trip bypass of the emergency diesel generators on an ECCS initiation signal during certain plant configurations.

Date of issuance: October 24, 1997

Effective date: Both units, as of date of issuance, to be implemented within 30.

Amendments Nos.: 221 and 226

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43373) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 19, 1997, as supplemented by letter dated August 25, 1997

Brief description of amendment: This amendment changes the Hope Creek Technical Specification (TS) 3.7.1.3, ''Ultimate Heat Sink,'' to raise the minimum allowable ulimate heat sink (UHS) water level from 76 feet to 80 feet, lower the maximum allowable UHS temperature from 88.6°F to 85°F, and reflect that continued plant operation to a UHS temperature of 87°F depends upon the association of UHS temperature and safety system availability. The associated Surveillance Requirement, TS 4.7.1.3, is changed to decrease the river water temperature, at which increasing temperature surveillance is required, from 85°F to 82°F. The requirements of TS 3.7.1.1, ''Safety Auxiliaries Cooling System (SACS),'' TS 3.7.1.2, ''Station Service Water System (SSWS),'' and TS 3.8.1.1, ''Electrical Power Systems,'' are revised to reflect the revised TS 3.7.1.3. In addition, the Bases for 3/4.7.1, ''Service Water Systems,'' are appropriately revised.

Date of issuance: October 28, 1997

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

Amendment No.: 106

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

Date of initial notice in **Federal Register:** June 18, 1997 (62 FR 33132) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 28, 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: March 31, 1997, as supplemented by letters dated July 16, August 26, and October 3, 1997

Brief description of amendment: This amendment changes Technical Specification (TS) 2.1.2, ''THERMAL POWER, High Pressure and High Flow,'' ACTION a.1.c for TS 3.4.1.1, ''Recirculation Loops,'' and the Bases for TS 2.1, ''Safety Limits.'' These changes are being made to implement an appropriately conservative Safety Limit Minimum Critical Power Ratio to include Cycle 8 specific analyses for all Hope Creek core and fuel designs.

Date of issuance: November 4, 1997

Effective date: The license amendment is effective as of its date of issuance and shall be implemented within 60 days.

Amendment No.: 107

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

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43374) The August 26 and October 3, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 4, 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: September 29, 1997

Brief description of amendment: This amendment changes Technical Specification (TS) 3/4.11.1, ''Liquid Effluent - Concentration.'' The change adds a requirement to perform weekly sampling and monthly and quarterly composite analyses of the Station Service Water System when the Reactor Auxiliaries Cooling System is contaminated. The licensee has also proposed an editorial change to TS Table 4.11.1.1.1-1. In Liquid Release Type B, the licensee is proposing that the acronym for Station Service Water System be changed from GSW to SSWS. This proposed change will be addressed in a future license amendment.

Date of issuance: November 6, 1997

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

Amendment No.: 108

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

Date of initial notice in **Federal Register:** October 6, 1997 (62 FR 52161) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 6, 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: September 24, 1997

Brief description of amendment: This amendment adds a Surveillance Requirement to Technical Specification 3/4.5.1, ''Emergency Core Cooling Systems'', to perform a monthly valve position verification for the four residual heat removal cross-tie valves.

Date of issuance: November 6, 1997

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

Amendment No.: 109

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

Date of initial notice in **Federal Register:** October 6, 1997 (62 FR 52162) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 6, 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: August 20, 1997

Brief description of amendment: This amendment changes the Technical Specifications (TSs) to provide for: 1) the relocation of suppression chamber volume references in Limiting Condition for Operation (LCO) 3.5.3 to the Hope Creek (HC) Updated Final Safety Analysis Report (UFSAR) and TS Bases as appropriate; 2) the revision of the suppression chamber volume currently listed in LCO 3.5.3.b; 3) the relocation of the suppression chamber volume references in LCO 3.6.2.1.a.1 to the UFSAR and TS Bases; and 4) the revision to the suppression chamber volume reference in TS 5.2.1 to reference the TS Bases section where this information will reside.

Date of issuance: November 6, 1997

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

Amendment No.: 110

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

Date of initial notice in **Federal Register:** September 24, 1997 (62 FR 50010) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 6, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 19, 1997, as supplemented September 29, 1997.

Brief description of amendment: The proposed amendment revises the Ginna Station Improved Technical Specifications by adding a note to the Containment Spray (CS) Limiting Condition for Operation 3.6.6 which allows the CS pumps in MODE 4 to be placed in pull-stop, and motor-operated valves 896A and 896B to have their DC control power restored with the valves placed in the closed position in order to perform interlock and valve testing of MOVs 857A, 857B, and 857C. A time limit of 2 hours is placed on this configuration for each test.

Date of issuance: October 29, 1997

Effective date: October 29, 1997

Amendment No.: 68

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

Date of initial notice in **Federal Register:** September 24, 1997 (62 FR 50011) The September 29, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 1997. No significant hazards consideration comments received: No.

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

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: September 17, 1997

Brief description of amendments: The amendments change Technical Specification 3/4.4.9, ''Specific Activity,'' and the associated Bases to reduce the limit associated with dose equivalent iodine-131. The steady-state dose equivalent iodine-131 limit would be reduced by 50 percent from 0.3 mu Curie/gram to 0.15 mu Curie/gram and the maximum instantaneous value would be reduced by 50 percent from 18 mu Curie/gram to 9 mu Curie/gram.

Date of issuance: October 29, 1997

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

Amendment Nos.: Unit 1 - 132; Unit 2 - 124

Facility Operating License Nos. NPF-2 and NPF-8: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** September 24, 1997 (62 FR 49998) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 29, 1997. No significant hazards consideration comments received: No.

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

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: August 14, 1997, as supplemented September 26 and October 1, 1997.

Brief description of amendment: This amendment changes the design basis as described in the Updated Safety Analysis Report by adding a description of the methodology utilized for determining the systems and components that are considered to require protection from tornado missiles.

Date of issuance: November 4, 1997

Effective date: November 4, 1997

Amendment No.: 90

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

Date of initial notice in **Federal Register:** September 16, 1997 (62 FR 48674). The September 26 and October 1, 1997, submittals provided supplemental information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 4, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: May 16, 1997 (TXX-97119)

Brief description of amendments: The amendments revised core safety limit curves and Overtemperature N-16 reactor trip setpoints based on analyses of the core configuration for CPSES Unit 2, Cycle 4. These changes apply equally to CPSES Units 1 and 2 licenses since the Technical Specifications are combined.

Date of issuance: October 30, 1997

Effective date: October 30, 1997

Amendment Nos.: Unit 1 - Amendment No. 55; Unit 2 - Amendment No. 41

Facility Operating License Nos. NPF-87 and NPF-89: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 16, 1997 (62 FR 38140) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 30, 1997. No significant hazards consideration comments received: No.

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

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

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By December 19, 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).

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

Date of application for amendment: October 24, 1997

Brief description of amendment: The amendment adds a footnote to Technical Specification 3.7.A.5, ''Primary Containment.'' The footnote provides a one time exception to the reverse flow testing requirement for containment isolation check valve 30-CK-432.

Date of issuance: October 30, 1997

Effective date: As of date of issuance and shall be implemented by November 2, 1997.

Amendment No.: 174

Facility Operating License No. DPR-35: This amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with the State of Massachusetts, and final no significant hazards consideration determination are contained in a Safety Evaluation dated October 30, 1997.

Attorney for licensee: W. S. Stowe, Esquire, Boston Edison Company, 800 Boylston Street, 36th Floor, Boston, Massachusetts 02199

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

NRC Project Director: Ronald B. Eaton, Acting Director

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

Date of amendment request: October 17, 1997

Brief description of amendment: The amendment revised Technical Specification 4.5.2b and associated Bases to eliminate the requirement to vent the centrifugal charging pump casings.

Date of issuance: November 3, 1997

Effective date: November 3, 1997

Amendment No.: 114

Facility Operating License No. NPF-42: The amendment revised the Technical Specifications. Press release issued requesting comments as to proposed no significant hazards consideration: Yes. October 24, 1997. Coffey County Today Newspaper (Kansas). Comments received: Yes. Comments were submitted by Mr. Dave Lochbaum of the Union of Concerned Scientists by letter dated October 29, 1997. Verbal comments were received from Larry Myers on October 28, 1997. The staff responded to these comments in the safety evaluation attached to the November 3, 1997, amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of Kansas and final determination of no significant hazards consideration are contained in a Safety Evaluation dated November 3, 1997.

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

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

NRC Project Director: William H. Bateman

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

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[FR Doc. 97–30217 Filed 11–18–97; 8:45 am]

BILLING CODE 7590–01–F

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–22883/812–10536]

EQ Advisors Trust; Notice of Application

November 12, 1997.

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

ACTION: Notice of Application for an order under (i) section 6(c) of the Investment Company Act of 1940 (the ''Act'') granting relief from sections 13(a)(2), 18(f)(1), 22(f), and 22(g) of the Act; and (ii) section 17(d) of the Act and rule 17d–1 to permit certain joint transactions.

Summary of Application: Applicants request an order to permit EQ Advisors Trust to implement a deferred compensation plan for certain of its trustees.

Applicants: HQ Advisors Trust (the ''Trust'') and EQ Financial Consultants, Inc. (the ''Manager'').

FILING DATES: The application was filed on April 7, 1997, and amendments were filed on July 14, 1997 and November 10, 1997.

Hearing or Notification of Hearing: An order granting the application will be issued unless the SEC orders a hearing. Interested persons may request a hearing by writing to the SEC's Secretary and serving applicant with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on December 8, 1997, and should be accompanied by proof of service on applicant, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 Fifth Street, N.W., Washington, D.C. 20549. EQ Advisors Trust, 1290 Avenue of the Americas, New York, New York 10104.

FOR FURTHER INFORMATION CONTACT: Deepak T. Pai, Staff Attorney, at (202) 942–0574, or Nadya B. Roytblat, Assistant Director, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee at the SEC's Public Reference Branch, 450 Fifth Street, N.W., Washington, D.C. 20549 (tel. 202–942–8090).

Applicants' Representations

1. The Trust is an open-end management investment company registered under the Act and organized as a Delaware business trust. The Trust is currently composed of several, separately managed series (''Portfolios''). The Trust offers shares in each of its Portfolios only to insurance companies and their separate accounts that fund variable annuity and variable life insurance contracts (''Variable Contracts''). The Trust is currently serving as the underlying investment medium for Variable Contracts issued by the Equitable Life Assurance Society of the United States (''Equitable''). The Trust may in the future offer its shares to separate accounts funding Variable Contracts of insurance companies unaffiliated with Equitable or directly to tax qualified pension and retirement plans outside the separate account context.

2. The Manager, an indirect wholly-owned subsidiary of Equitable, has overall responsibility for the investment management and administration of the Trust and its Portfolios. Rowe Price-Fleming International, Inc., T. Rowe Price Associates, Inc., Putnam Investment Management, Inc., Massachusetts Financial Services Company, Morgan Stanley Asset Management, Inc., Warburg Pincus Counsellors, Inc., and Merrill Lynch Asset Management, L.P. serve as the sub-advisers (each an ''Adviser'') to one or more Portfolios. Applicants request that the relief apply to the Trust and any registered open-end management investment company that in the future is advised by the Manager or any entity controlled by the Manager.[1]

3. The Trust's board of trustees (''Trustees'') currently consists of six members, two of whom are ''interested persons'' of the Trust within the meaning of Section 2(a)(19) of the Act. The four non-interested Trustees (''Eligible Trustees'') will receive an annual retainer fee, a fee for each board meeting and committee meeting attended, and an additional fee for performing special services for the Trust.

4. The deferred compensation plan for Eligible Trustees (the ''Plan'') was ratified by the Trustees on March 31, 1997. The purpose of the Plan is to permit Eligible Trustees to defer receipt of all or a portion of their fees to enable them to defer payment of income taxes, to avoid a loss or reduction of Social Security benefits, or for other reasons. Applicants believe that the Plan will better enable the Trust to attract and retain high caliber trustees. The Plan may be amended from time to time, provided that, any amendments are not inconsistent with the relief granted pursuant to this application.

5. Under the Plan, each Eligible Trustee who elects to defer receipt of

[1] The Manager is an investment adviser to the Trust and serves in such capacity pursuant to a contract subject to section 15 of the Act. All registered open-end investment companies that currently intend to rely on the order have been named as applicants. Any other existing or future investment company that relies on the order will comply with the terms and conditions of the order.