(Public Meeting) (if needed)
2:00 p.m.—Briefing on Salem (Public Meeting), (Contact: John Zwolinski, 301–415–1453)

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.

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The NRC Commission Meeting Schedule can be found on the Internet at: http://www.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, D.C. 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.

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Dated: May 30, 1997.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 97–14679 Filed 6–2–97; 10:21 am]

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NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 12, 1997, through May 22, 1997. The last biweekly notice was published on May 21, 1997 (62 FR 27792).

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

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

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

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

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

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

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, 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.

Consumers Power Company, Docket No. 50-155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: April 22, 1997 (supersedes October 15, 1996, request)

Description of amendment request: The proposed amendment would revise the Big Rock Point Technical Specifications to correct several administrative and editorial inconsistencies.

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

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

The proposed changes are clarifications within the Technical Specifications, and do not alter the technical content of the technical specifications. Plant operation or configuration is not affected. The postulated doses received by the general public and plant personnel as a direct result of accidents previously described, are not affected. Plant operation or configuration is not affected. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes are either clarifications to correct inconsistencies within the Technical Specifications, or corrections of typographical errors. The proposed changes do not alter the facility in any way, therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change[s] [do] not affect any margin of safety as defined by 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 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: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770

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

NRC Project Director: John N. Hannon

Consumers Power Company, Docket No. 50-155, Big Rock Point Plant, Charlevoix County, Michigan

Date of amendment request: April 30, 1997

Description of amendment request: The proposed amendment would alter the company name in the Facility Operating License DPR-6 and Technical Specifications for the Big Rock Point Plant. Specifically, the proposed amendment would revise the company name from ''Consumers Power Company'' to ''Consumers Energy Company.''

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

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

The proposed changes alter the company name in the Facility Operating License and Technical Specifications to reflect the change from ''Consumers Power Company'' to ''Consumers Energy Company''. The company will continue to own all of the same assets, will continue to serve the same customers, and will continue to honor all existing obligations and commitments.

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter the design, function, or operation of any plant structure, system, or component.

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

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

The proposed changes alter the company name in the Facility Operating License and Technical Specifications to reflect the change from ''Consumers Power Company'' to ''Consumers Energy Company''. The company will continue to own all of the same assets, will continue to serve the same customers, and will continue to honor all existing obligations and commitments.

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter the design, function, or operation of any plant structure, system, or component.

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

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

Since the proposed changes do not alter the technical content of any Facility Operating License or Technical Specifications requirements, they do not alter the design, function, or operation of any plant structure, system, or component.

Therefore, the changes will 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: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770

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

NRC Project Director: John N. Hannon

Duke Power Company, Docket No. 50-413, Catawba Nuclear Station, Unit 1, York County, South Carolina

Date of amendment request: May 8, 1997

Description of amendment request: The proposed amendment would add a phrase to the footnote to Section 3.4.1.2 of the Technical Specifications that would permit all reactor coolant pumps (RCPs) to be deenergized for up to 4 hours during Mode 3 on a one-time basis. Currently, the RCPs are permitted to be deenergized for up to 1 hour during Mode 3. The proposed change would allow the licensee to perform a natural circulation test using the new steam generators (installed in late 1996).

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

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

The proposed natural circulation test would be performed in Mode 3 with the reactor subcritical. This transient is bounded by the transient analyzed in UFSAR [Updated Final Safety Analysis Report] Section 15.2.6, Loss of Non-Emergency AC Power to the Station Auxiliaries. For this ANS [American Nuclear Society] Condition II event, the reactor is assumed to be operating at 102% power, the turbine driven auxiliary feedwater pump is assumed unavailable and each steam generator is assumed to have 18% of the steam generator tubes plugged. By contrast, the planned natural circulation test would be performed with the reactor subcritical, less than 0.1% of the tubes plugged in each steam generator, and all support systems such as auxiliary feedwater, operable for the test. Therefore, the proposed natural circulation test would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) The activity does not create the possibility of a new or different type of accident from any accident previously evaluated.

The proposed change does not involve a physical alteration of the unit (i.e., no new or different equipment will be installed), nor will the function of equipment be changed. The change will allow for a one time performance of a natural circulation test in Mode 3 which will provide useful data on the natural circulation capabilities of the new Babcock and Wilcox International (BWI) steam generators that were recently installed at Catawba Unit 1. The test data will be utilized to validate analysis and simulator models. Plant operators will also receive valuable experience from performance of the test. The test will be conducted using written and approved procedures. An Emergency procedure (EP/1/A/5000/ECA-0.1) is also available to the Operators for this test. This test is bounded by the Loss of Non-Emergency AC Power to the Station Auxiliaries event in Section 15.2.6 of the Catawba UFSAR. For these reasons, the planned natural circulation test will not create the possibility of a new or different type of accident from any previously evaluated.

3) The activity does not involve a significant reduction in the margin of safety.

Margin of safety is associated with confidence in the ability of the fission product barriers (the fuel and fuel cladding, the Reactor Coolant System pressure boundary, and the containment) to limit the level of radiation doses to the public. As demonstrated by the bounding UFSAR analysis in Section 15.2.6, none of the fission product barriers are adversely impacted by the proposed one-time change. The proposed change does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. For these reasons, the activity 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 proposed amendments involve no significant hazards consideration.

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: February 24, 1997, as supplemented on April 24, 1997.

Description of amendment request: The licensee proposed changes to Technical Specification Section 6.9.1.7, Core Operating Limits Report, to reflect use of the Westinghouse Best Estimate Large Break Loss-of-Coolant Accident (LOCA) methodology for large break LOCA analysis, including supporting documents.

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:

Question 1 Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a Loss of Coolant Accident

(LOCA). The consequences of a LOCA are not being increased. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46 paragraph (b). No other accident is potentially affected by this change. Therefore, neither the BiWeekly probability nor the consequences of an accident previously evaluated is increased due to the proposed change.

Question 2 Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of existing plant equipment. All plant systems will perform as designed in response to a potential accident. Therefore, the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Question 3 Does the proposed amendment involve a significant reduction in the margin of safety?

The analysis in support of the proposed license amendment realistically models the expected response of the Turkey Point Units 3 & 4 nuclear core during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different break locations and other variations in properties have been calculated to provide assurance that the most severe postulated loss of coolant accidents were analyzed. It has been shown by the analysis that there is a high level of probability the criteria contained in 10 CFR 50.46 paragraph (b) would not be exceeded. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: February 17, 1997 as revised May 1, 1997.

Description of amendment request: The proposed amendment would change the Crystal River Unit 3 (CR-3) Technical Specifications (TS) to implement 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Reactors,'' Option B. This option allows to change from prescriptive testing requirements to performance-based testing requirements based on the leakage rate testing history of the containment and components. The proposed TS changes include revision to TS 3.6.1, 3.6.3, and addition of ''Containment Leakage Rate Testing Program'' to TS 5.0. The licensee did not propose any deviations from methods approved by the Commission and endorsed in the applicable regulatory guide. This notice supersedes the previous notice dated February 28, 1997 (62 FR 9214)

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

Criterion 1

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

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

The proposed changes to the TS are to implement Option B of 10 CFR 50, Appendix J, at CR-3. The proposed changes will result in increased intervals between containment leakage tests based on the leakage rate testing history. The proposed changes do not involve a change to the plant design or operation and does not change the testing methodology.

NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' provides the technical basis of 10 CFR 50, Appendix J, Option B. NUREG-1493 contains a detailed evaluation of the expected leakage from containment and the associated consequences. The increased risk due to increasing the intervals between containment leakage tests was also evaluated. The NUREG used a statistical approach to determine that the increase in the expected dose to the public due to decreasing the testing frequency is extremely low. NUREG-1493 also concluded that a small increase is justifiable in comparison to the benefits from decreasing the testing frequency. The primary benefit is in the reduction in occupational radiation exposure.

Criterion 2

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

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

The proposed TS amendment incorporates the performance-based testing approach authorized by 10 CFR 50 Appendix, J, Option B. Decreasing the testing frequency allowed by this change does not involve a change to plant design or operation. Safety related equipment and safety functions are not altered as a result of this change. Decreasing the testing frequency does not affect testing methodology. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to the initiation of any accidents.

Criterion 3

Does not involve a significant reduction in the margin of safety.

This TS amendment does not involve a significant reduction in the margin of safety.

The proposed TS amendment does not change the methodology of the containment leakage rate testing program or program acceptance criteria. The proposed TS change does affect the frequency of containment leakage rate testing. With an increased interval between tests, a small possibility exists that an increase in leakage could go undetected for a longer period of time. Based on the operational experience at CR-3, it has been demonstrated that the leak-tightness of the containment building has consistently been significantly below the allowable leakage limit. Adequate controls are in place to ensure that required maintenance and modifications are performed.

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

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

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: March 27, 1997, as supplemented April 3, and May 1, 1997.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) for the Crystal River Nuclear Plant Unit 3 (CR3) relating to the Once Through Steam Generator's (OTSG's) tube inspection acceptance criteria. Specifically, the licensee proposed to:

(1) revise TS 3.4.12 (d) to specify 150 gallons per day limit on primary-to-secondary leakage through either OTSG;

(2) add TS 5.6.2.10.2 e. to define inspection requirements and disposition criteria for applicable tubes in the ''B'' OTSG first span;

(3) revise TS 5.6.2.10.4.a.7 to define ''pit-like Intergranular attack indications≥

(4) revise TS 5.6.2.10 and 5.7.2 to delete requirements that were specific to the interim tube plugging criteria applicable until Refuel 11.

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

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

FPC Response:

No. The CR-3 components addressed by this proposed change are the Once Through Steam Generators (OTSGs), identified by plant tagging procedures as RCSG-1A and RCSG-1B. The OTSGs are straight tube, straight shell heat exchangers which allow for heat removal and the subsequent production of steam as a result of heat transfer from the primary side reactor coolant to the secondary side feedwater. Proposed changes are; retaining reduced primary-to-secondary leak rates approved previously for one cycle only, returning inspection result reporting requirements to those previously implemented, and establishing new inspection requirements for the ''B'' OTSG. Sunset clauses are being removed from pages containing requirements effective for one refueling outage and subsequent operating cycle only.

Based on review of Chapter 14 of the CR-3 Final Safety Analysis Report (FSAR), FPC performed analyses to assess the consequences of a steam generator tube rupture event, including the complete severing of a steam generator tube. The analyses concluded that CR-3 was sufficiently designed to ensure that, in the event of a steam generator tube rupture, the radiological doses would not exceed the allowable limits prescribed by 10 CFR 100, and would not result in additional tube failures and further degradation of the reactor coolant pressure boundary.

Retaining the present primary-to-secondary leakage limit (LCO 3.4.12, RCS Operational Leakage) that was previously approved for the current operating cycle will continue to provide assurance that should a significant leak occur, it would be detected and the plant will be shut down in a timely manner to reduce the likelihood of a potential tube rupture. This value of primary-to-secondary leakage applicable to both OTSGs is conservative relative to existing safety analyses and would result in lower doses than currently calculated and found acceptable. Removing reporting requirements specific to use of alternate flaw sizing criteria approved for Refueling Outage 10 only, and returning to previous reporting requirements applicable to both OTSGs, has no effect on operating plant safety. These requirements are administrative only and do not affect steam generator inspection or disposition of inspection results.

The proposed change to the ''B'' OTSG inspection criteria establishes that future inspections will include 100% inspection of the first span of specific tubes which are known to have indications of degradation. The degradation of these tubes is attributed to a common non-random mechanism.

The results of inspections of these tubes will be dispositioned using the same criteria as all other OTSG tubes for determination of the need for plugging or sleeving. Therefore, the proposed change will not increase the probability or consequence of an accident previously evaluated as all tubes degraded beyond acceptable limits will be subject to consistent corrective actions.

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

FPC Response:

No. The purpose of OTSG tube inspection is to identify tubes that may have a higher potential for failure due to degradation that results in a reduced ability to withstand operating conditions. Neither the type of inspection of OTSG tubes nor the process for performing inspections will be changed by this amendment. Consistent criteria will be applied to disposition inspection results and consistent corrective actions will be taken for tubes that exceed this criteria. Retaining the lower leakage limit is conservative relative to existing analyses. Changes to revise requirements for reporting inspection results, and remove ''sunset'' clauses addressing the applicability of License Amendment 154 until Refueling Outage 11 only, do not alter the design or operation of the OTSGs. Therefore, no new or different kind of accident will be created as a result of these changes.

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

FPC Response:

No. The analyses that have been performed on the effects of OTSG tube failures, as reported in the CR-3 FSAR, have demonstrated that internal and offsite consequences are within allowable limits. This change will not alter the acceptance criteria for inspection results. Since this change will assure that a group of tubes with existing first span pit-like inter-granular attack indications are inspected each inspection period, the likelihood of detecting active degradation, as well as the probability of repairing degraded tubes prior to the degradation resulting in a through-wall opening or tube rupture, is increased. Retaining the currently accepted primary-to-secondary leakage limit continues to provide assurance that should a significant leak occur, it would be detected and the plant will be shut down in a timely manner to reduce the likelihood of a potential tube rupture, thereby maintaining or improving the existing margin of safety. Changes to revise requirements for reporting inspection results, and remove ''sunset'' clauses addressing the applicability of License Amendment 154 until Refueling Outage 11 only, do not alter the design or operation of the OTSGs. Therefore, these changes 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: May 8, 1997

Description of amendment request: The proposed amendment incorporates additional analytical methods, GPU Nuclear Topical Reports, TR-078, TR-087, TR-091, and TR-092P, previously approved by the NRC, to Technical Specifications (TS) Section 6.9.5.2. These Topical Reports will be utilized by GPU Nuclear to perform core reload design analysis for the Three Mile Island, Unit 1 (TMI-1) Facility. TS 6.9.5.2 is also being editorially revised to relocate the existing note that the current revision level shall be specified in the Core Operating Limits Report (COLR) such that it applies to the additional Topical Reports, as well as BAW-10179 P-A.

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

GPU Nuclear has determined that this Technical Specification Change Request poses no significant hazards as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed change to reference the analytical methodologies specified in GPU Nuclear Topical Reports TR-078, TR-087, TR-091,and TR-092 use[d] in TMI-1 core reload design analysis is considered administrative since these Topical Reports

have been reviewed and approved by the NRC for use at TMI-1.

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change to reference NRC-approved GPU Nuclear Topical Reports TR-078, TR-087, TR-091, and TR-092P will continue to ensure that approved methods and criteria are used to establish core operating limits.

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed change to reference NRC-approved GPU Nuclear Topical Reports TR-078, TR-087, TR-091, and TR-092P maintains existing margins of safety since approved methods and criteria are still used to establish core operating limits.

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

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

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

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

NRC Project Director: Patrick D. Milano, Acting

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

Date of amendment request: May 8, 1997

Description of amendment request: The proposed amendment would modify the minimum accuracy stated in Technical Specification (TS) Table 3.3-8, ≥Meteorological Monitoring Instrumentation,'' for the instruments used to measure wind speed and air temperature - delta T. TS Bases Section 3/4.3.3.4 would also be modified to reflect the proposed changes to TS Table 3.3-8.

Regulatory Guide 1.23 (Safety Guide 23), ''Onsite Meteorological Programs,'' dated March 17, 1972, provides recommended instrument accuracies for meteorological instrumentation. The proposed minimum instrument accuracies for the air temperature - delta T and the wind speed (only when the wind speed is greater than 5 miles per hour) do not meet the recommended accuracies of Regulatory Guide 1.23. However, margin is included to account for uncertainties.

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

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

The proposed changes modify the accuracy requirements for the instruments which are used to measure wind speed and air temperature - delta T. The data obtained from the meteorological instrumentation would be used to: a) estimate the public dose following routine or accidental releases of airborne radioactivity, b) make decisions regarding actions to protect the public in the event of an accident involving a release of airborne radioactivity, and c) establish radiological dispersion parameters to determine radiological doses in design basis accident calculations.

The proposed minimum instrument accuracy requirements are more than sufficient to meet the purposes denoted above. The meteorological parameters measurement uncertainties insignificantly affect the results when compared to the accuracies of the source term estimates, meteorological dispersion models, dose models, and meteorological forecasting. Therefore, there is no impact on the consequences (offsite doses) associated with previously evaluated accidents.

The proposed changes do not alter the way any structure, system, or component functions, do not alter the manner in which the plant is operated, and do not have any impact on the protective boundaries and safety limits for the protective boundaries. Therefore, the proposed changes do not impact the probability of any previously evaluated accidents.

Thus, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequence of an accident previously evaluated.

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

The proposed changes modify the accuracy requirements for the instruments which are used to measure wind speed and air temperature - delta T. The data provided by these instruments assist in responding to a design basis accident which may involve a release of airborne radioactivity. The instruments are used for post accident monitoring and serve a passive role; they cannot initiate or mitigate any accident.

The proposed changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. They do not introduce any new failure modes.

Thus, the license amendment request does not create the possibility of a new or different kind of accident from any previously analyzed.

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

As discussed above, the proposed changes modify the accuracy requirements for the instruments which are used to measure wind speed and air temperature - delta T which could impact the radiological dispersion coefficient used to determine radiological doses in design basis accident calculations. However, the differences in the instrument accuracies and the Regulatory Guide 1.23 requirements have been determined not to significantly affect the dispersion coefficients. Thus, there is no significant impact on offsite doses associated with previously analyzed accidents. Therefore, there is no significant reduction in the margin of safety for the design basis accident analysis.

Thus, the license amendment request does not involve a significant reduction in the margin of safety.

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: April 14, 1997

Description of amendment request: Technical Specification 3.4.9.3.a requires two relief valves be operable to protect the reactor coolant system from overpressurization when any reactor coolant system cold leg is less than 350—F. The proposed amendment revises the setpoint of the residual heat removal suction relief valves.

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

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

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

The proposed change to Technical Specification 3.4.9.3 to decrease the setpoint of the residual heat removal suction relief valves from 450 psig [plus or minus] 3% to 440 psig [plus or minus] 3% ([greater than or equal to] 426.8 psig and [less than or equal to] 453.2 psig) is consistent with the design capabilities and system requirements of the relief valves and the relief valves are not credited in previously evaluated accidents.

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

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

The proposed change to Technical Specification 3.4.9.3 to decrease the setpoint of the residual heat removal suction relief valves from 450 psig [plus or minus] 3% to 440 psig [plus or minus] 3% ([greater than or equal to] 426.8 psig and [less than or equal to] 453.2 psig) does not change the operation of the residual heat removal system, reactor coolant system or any system component during normal or accident evaluations. The proposed change to the setpoint of the residual heat removal suction relief valves from 450 psig [plus or minus] 3% to 440 psig [plus or minus] 3% ([greater than or equal to] 426.8 psig and [less than or equal to] 453.2 psig) also ensures protection of the reactor coolant system against cold overpressurization transients in accordance with the requirements of 10CFR50, Appendix G.

Therefore, the proposed change 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 change to Technical Specification 3.4.9.3 to decrease the setpoint of the residual heat removal suction relief valves from 450 psig [plus or minus] 3% to 440 psig [plus or minus] 3% ([greater than or equal to] 426.8 psig and [less than or equal to] 453.2 psig) provides an acceptable allowance between the maximum relief valve setpoint ([less than or equal to] 453.2 psig) and 10CFR50, Appendix G requirements. The proposed change to the setpoint provides sufficient allowance between the minimum relief valve setpoint ([greater than or equal to] 426.8 psig) and reactor coolant system pressure when residual heat removal system is unisolated from the reactor coolant system to minimize the probability of an inadvertent residual heat removal system relief valve opening.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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: April 28, 1997

Description of amendment request: Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1, 4.5.2.f, and 4.5.2.h require the charging and safety injection pumps to be tested on a periodic basis and after modifications that alter subsystem flow characteristics. The proposed amendment would increase the required differential pressure at recirculation flow for the safety injection and centrifugal charging pumps; decrease the required individual safety injection and centrifugal charging pump injection line flow rate; increase the allowed individual safety injection pump total flow rate; and make editorial changes.

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

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and has concluded that the changes do 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 changes do not involve [an] SHC because the changes would not:

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

The proposed changes to Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1, and 4.5.2.f to increase the required discharge pressure for the centrifugal charging pumps on recirculation flow during surveillance testing from [greater than or equal to] 2411 psid to [greater than or equal to] 5676 ft (2464 psid) are consistent with centrifugal charging pump design requirements. The change in the referenced units from differential pressure measured in psid to total head measured in feet for the centrifugal charging pumps and safety injection pumps during surveillance testing is an administrative change.

The proposed changes to Technical Specification Surveillance 4.5.2.f to increase the required discharge pressure for the safety injection pumps on recirculation flow during surveillance testing from [greater than or equal to] 1348 psid to [greater than or equal to] 3240 ft (1406 psid) are consistent with safety injection pump design requirements.

The proposed changes to Surveillance 4.5.2.h: to decrease the required individual centrifugal charging pump injection line flow rate sum from [greater than or equal to] 339 gpm to [greater than or equal to] 310.5 gpm, decrease the required individual safety injection pump injection line flow rate sum from [greater than or equal to] 442.5 gpm to [greater than or equal to] 423.4 gpm, increase the required individual safety injection Pump A total flow rate from [less than or equal to] 670 gpm to [less than or equal to] 675 gpm, and increase the required individual safety injection Pump B total flow rate from [less than or equal to]

650 gpm to [less than or equal to] 675 gpm are consistent with centrifugal charging pump and safety injection pump design requirements.

The proposed changes are consistent with equipment design requirements and performing surveillance testing does not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes to the surveillance testing of the centrifugal charging pumps and safety injection pumps provide the necessary assurance that the pumps will function consistent with the flows used in the accident analyses and does not involve a significant increase in the consequence of an accident previously evaluated.

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

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

The proposed changes to the surveillance testing of the centrifugal charging pumps and safety injection pumps do not change the operation of the centrifugal charging or safety injection systems or any of its components during normal or accident evaluations. The increase in the allowed maximum safety injection pump flow does not impact the cold overpressure accident analysis.

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

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

The proposed changes to Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1 and 4.5.2.f to increase the required discharge pressure for the centrifugal charging pumps on recirculation flow during surveillance testing from [greater than or equal to] 2411 psid to [greater than or equal to] 5676 ft (2464 psid) provides an acceptable margin between the required surveillance and design pump performance to provide assurance that the pumps will operate consistent with the assumptions of the accident analysis.

The proposed changes to Technical Specification Surveillance 4.5.2.f to increase the required discharge pressure for the safety injection pumps on recirculation flow during surveillance testing from [greater than or equal to] 1348 psid to [greater than or equal to] 3240 ft (1406 psid) provides an acceptable margin between the required surveillance and design pump performance to provide assurance that the safety injection pumps will operate consistent with the assumptions of the accident analysis.

The proposed changes to Surveillance 4.5.2.h to decrease the required individual centrifugal charging pump injection line flow rate sum from [greater than or equal to] 339 gpm to [greater than or equal to] 310.5 gpm, decrease the required individual safety injection pump injection line flow rate sum from [greater than or equal to] 442.5 gpm to [greater than or equal to] 423.4 gpm, increase the required individual safety injection Pump A total flow rate from [less than or equal to] 670 gpm to [less than or equal to] 675 gpm and increase the required individual safety injection Pump B total flow rate from [less than or equal to] 650 gpm to [less than or equal to] 675 gpm are consistent with the assumptions of the accident analysis. The maximum allowed safety injection flow is consistent with the vendor recommendation for maximum continuous runout flow. Also, the safety injection

pumps are disabled during specific normal operating modes, consistent with the assumptions of the accident analysis, to ensure that they can not be an injection source when the cold overpressure system is required to be operable and thus the increase in maximum safety injection pump flow does not affect the cold overpressure accident analysis.

The change in the referenced units in Technical Specification Surveillances 4.1.2.3.1, 4.1.2.4.1 and 4.5.2.f from differential pressure measured in psid to total head measured in feet for the centrifugal charging pumps and safety injection pumps during surveillance testing is an administrative change.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: April 28, 1997

Description of amendment request: Technical Specification 3.7.6 requires that flood protection be provided for the service water pump cubicles and components when the water level exceeds a specific value. The proposed amendment (1) adds the closing of the service water pump cubicle sump drain valves, (2) revises the wording of the action statement to be consistent with the limiting condition for operation, and (3) revises the associated Bases section.

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

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

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

The proposed changes to Technical Specification 3.7.6 identify additional manual actions to be performed to provide external lood protection for the service water pump cubicles in the event of high water level (13 ft MSL) [mean sea level]. The cubicle sump drain valves which are to be closed are part of a modification which installed a drain line from the sump of each cubicle to the intake bay in order to provide a passive means of removing internal leakage from the cubicle. The cubicle sump drain valves are normally maintained in the open position.

The drain valves meet the intent of RG [Regulatory Guide] 1.59 for ''hardened protection'' and RG 1.102 for ''incorporated barriers'' in a manner similar to that of the cubicle watertight doors. RG 1.59 states that hardened protection ''must be passive and in place, as it is to be used for flood protection, during normal plant operation''. RG 1.102 states that ''the plant should be designed and operated to keep doors necessary for flood protection closed during normal operation''. The Response to FSAR [Final Safety Analysis Report] Question No. 240.9 established the acceptability of the practice of maintaining one service water pump cubicle watertight door open and the other door closed during normal operations.

The proposed change in the action statement to initiate action when water level is exceeding 13 feet MSL rather than at 13 feet MSL is a clarification only which provides consistency between the limiting condition for operation and the action statements.

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

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

The proposed changes to Technical Specification 3.7.6 identify additional, simple to perform manual actions to provide external flood protection for the service water pump cubicles.

The proposed change in the action statement to initiate action when water level is exceeding 13 feet MSL rather than at 13 feet MSL and the proposed changes to the bases are considered clarifications.

Therefore, the proposed change 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 Technical Specification 3.7.6 identify additional, simple to perform manual actions to provide external

flood protection for the service water pump cubicles in the event of high water level (13 ft MSL). The plant modification which made these additional actions necessary was made to provide for improved internal flood protection.

The proposed change in the action statement to initiate action when water level is exceeding 13 feet MSL rather than at 13 feet MSL and the proposed changes to the bases are considered clarifications.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: May 1, 1997

Description of amendment request: Technical Specifications 3/4.8.2.2 and 3/4.8.3.2 specify which electrical power systems are required to be operable in Modes 5 and 6. The proposed amendment would clarify the requirements by identifying the specific equipment required and their alignments in Modes 5 and 6.

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

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

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

The proposed change to Technical Specification 3/4.8.2.2 to replace the wording ''As a minimum, one 125 volt battery bank and its associated full capacity charger'' to ''As a minimum, one Train (A or B) of batteries and their associated full capacity chargers'' will increase the required battery banks operable from one to two.[≥]

This change is being proposed to resolve an inconsistency with Technical Specification 3/4.8.3.2 which currently requires two battery banks energized in modes 5 and 6.

The proposed change to...Technical Specifications 3/4.8.2.2 and 3/4.8.3.2 to identify the specific equipment required and its alignment during modes 5 and 6 is being proposed to reduce the vagueness in the present Technical Specifications. This proposed change will specify the equipment required operable for the electrical distribution systems during modes 5 and 6.

These proposed changes are considered administrative and do not alter the manner in which any system or component is operated or expected to respond during an accident. Therefore, the proposed changes do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to Technical Specification 3/4.8.2.2 to increase the required battery banks operable from one to two and to reword Technical Specifications 3/4.8.2.2 and 3/4.8.3.2 to identify the specific equipment required operable during modes 5 and 6 do not alter the manner in which any system or component is operated or expected to respond during normal or accident conditions.

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

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

The proposed changes to Technical Specification 3/4.8.2.2 to increase the required battery banks operable from one to two is being proposed to resolve an inconsistency with Technical Specification 3/4.8.3.2 which currently requires two battery banks energized in modes 5 and 6. This is considered an administrative change.

The proposed changes to...Technical Specifications 3/4.8.2.2 and 3/4.8.3.2 are being proposed to reduce the vagueness in the present technical specifications by identifying the specific equipment required operable during modes 5 and 6. The change will provide a greater level of assurance that the electrical distribution systems will be correctly aligned and surveilled. This is also considered an administrative change.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: May 5, 1997

Description of amendment request: Technical Specification Surveillance 4.8.4.1 requires periodic testing of lower voltage circuit breakers for all containment penetration conductor overcurrent protective devices. The proposed amendment would modify the requirements for determining the operability of lower voltage circuit breakers by using the manufacturer's curve of current versus time to test delay trip elements, clarify the use of two pole in series testing, and expand the Bases description of the testing.

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

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The bases 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 change to Technical Specification Surveillance 4.8.4.1 to modify the requirements for determining the operability of lower voltage circuit breakers by using the manufacture's curve of current versus time to test long time and short-time delay trip elements will not change the requirement that periodic testing be performed to determine breaker operability. The circuit breaker testing is consistent with the design of the components and performing surveillance testing does not involve a significant increase in the probability of an accident previously evaluated. The proposed change will provide assurance that the breakers will perform consistent with accident analyses and does not involve a significant increase in the consequence of an accident previously evaluated.

The proposed change to the surveillance to modify the wording associated with the use of two pole in series testing to determine Molded Case Circuit Breaker (MCCB) operability following the failure of [an] MCCB to pass a single pole test was previously approved in License Amendment No. 13. The modified wording clarifies the testing by specifically stating in the surveillance that the two pole in series test determines MCCB operability. This is considered an administrative change.

The proposed change to expand the description of the long-time and short-time delay trip elements testing in the Bases Section is also considered an administrative change.

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

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

The proposed change to use a curve of current versus time instead of the description in Technical Specification Surveillance 4.8.4.1 of the [] long-time and short-time delay trip element testing does not alter the design, operation, or maintenance of the lower voltage circuit breakers.

The proposed change to the surveillance to modify the wording associated with the use of two pole in series testing to determine MCCB operability and the expanded description of the long-time and short-time delay elements testing in the Bases Section are considered administrative changes.

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

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

The current wording of Technical Specification Surveillance 4.8.4.1 requires testing of long-time delay trip elements with a current value of exactly 300% of the pickup setting and short-time delay trip elements with a current value of exactly 150% of the pickup setting. The testing [cannot] be performed at exact values. Circuit breaker manufactures develop a curve of current versus time for each breaker type that specifies the allowable time to trip for a specified current. Using the curve for a given breaker type, the operability of a circuit breaker can be verified by inserting a given current and verifying that the breaker trips within the allowable time delay band width for that current. Testing by the industry is typically performed at approximately 300% of the pickup setting for long-time delay trip elements and approximately 150% of the pickup setting for short-time delay trip elements. The proposed change to the surveillance to modify the requirements for determining the operability of circuit breakers by using the manufacturer's curve of current versus time to test delay trip elements will continue to provide assurance that lower voltage circuit breakers for all containment penetration conductor overcurrent protective devices will operate consistent with the assumptions of the accident analysis.

The proposed change to the surveillance to modify the wording associated with the use of two pole in series testing to determine MCCB operability and the expanded description of the long-time and short-time delay trip elements testing in the Bases Section are considered administrative changes.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: May 5, 1997

Description of amendment request: Technical Specification Surveillance 4.5.2.b.1 requires that the emergency core cooling system (ECCS) piping be verified full of water at least once per 31 days. The proposed amendment would revise the surveillance to exempt the operating charging pump(s) and associated piping from the requirement to be verified full of water and move the description of the verification method from the surveillance to the Bases section.

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

NNECO has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The bases 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 change to Technical Specification Surveillance 4.5.2.b.1 to exempt the operating centrifugal charging pump(s) and associated piping from the requirement to be vented will not effect the requirement the ECCS piping be full of water. An operating centrifugal charging pump and the associated piping is self venting and cannot develop voids and pockets of entrained gases. This change is consistent with the design of the charging system and ensuring that ECCS piping is full of water does not involve a significant increase in the probability or consequence of an accident previously evaluated.

The proposed change Technical Specification Surveillance 4.5.2.b.1 to move and expand the description of the venting method from the surveillance to the Bases Section are considered administrative changes.

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

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

The proposed change to exempt the operating centrifugal charging pump(s) and associated piping from the requirement to be periodically vented by crediting its self venting capabilities does not change the operation of the charging system or any of its components during normal or accident evaluations.

The proposed changes to move and expand the description of the venting method from the surveillance to the Bases Section are considered administrative changes.

Therefore, the proposed change 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 change to Technical Specification Surveillance 4.5.2.b.1 to exempt the operating centrifugal charging pump(s) and associated piping from the requirement to be manually vented by crediting its self venting capabilities, is consistent with the design of the charging system. This proposed change continues to ensure that ECCS piping is full of water and thus, does not involve a significant reduction in a margin of safety.

The proposed change to Technical Specification Surveillance 4.5.2.b.1 to move the description of the venting method from thesurveillance to the Bases Section is considered an administrative change. Currently the surveillance identifies that ECCS piping is to be verified full of water by venting ECCS pump casings and accessible discharge piping high points except for the RSS [recirculation spray system] pump, RSS heat exchanger and associated RSS piping that are not maintained filled with water during plant operation. The venting description will be expanded when moved to the bases to include an exclusion for the above described operating centrifugal charging pump(s) and associated piping and the venting method used for nonoperating centrifugal charging pumps. The centrifugal charging pumps have top mounted suction and discharge nozzles and do not have casing vents. The pump manufacturer has indicated that venting the pump suction pipe will assure that the pump is full of water. This venting of the nonoperating centrifugal charging pumps is accomplished by using a pump suction line test connection.

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

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

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

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

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

NRC Deputy Director: Phillip F. McKee

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

Date of amendment request: April 17, 1997

Description of amendment request: This license amendment request revises Technical Specification (TS) 2.12, ''Control Room System,'' to delete the Limiting Conditions of Operation (LCO) and associated surveillance for the control room temperature and replace it with an LCO and surveillance on the control room air conditioning (A/C) system. The remainder of TS 2.12 is being rewritten consistent with the requirements of the Combustion Engineering Standard TS (NUREG-1432, Rev. 1). In reviewing requirements for refueling and shutdown operations, additional TS improvement were identified. Therefore, the definition section, TS 2.1 ''Reactor Coolant System,'' 2.6 ''Containment System,'' 2.8 ''Refueling Operations,'' and associated surveillance requirements are proposed for revision to incorporate the design basis requirements for refueling operations and to correspond to NUREG-1432.

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

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

The proposed change will incorporate new requirements for the control room air conditioning system, control room filtration system, and refueling operations. In addition, the proposed change will ensure that the Limiting Condition for Operations and surveillance requirements are consistent with the design basis of a fuel handling accident as documented in the FCS Updated Safety Analysis Report (USAR).

CONTROL ROOM SYSTEMS

The control room air conditioning (A/C) system consists of two redundant A/C units, VA-46A and VA-46B. Each unit has sufficient capacity to meet the cooling requirements for personnel and equipment inside the control room envelope. Each A/C unit is equipped with an air-cooled condenser located inside a protective enclosure outdoors on the roof of the Auxiliary Building. Each A/C unit's refrigerant compressor, air cooling coils, fans, and dampers are located inside of the control room envelope. Each unit has a waterside economizer coil that allows air cooling with Component Cooling Water (CCW). When cooling water temperature is sufficiently low, a temperature-activated valve at each A/C unit allows cooling water flow through the waterside economizer. This valve also diverts flow away from the waterside economizer if cooling water temperature is too high. The air-operated CCW isolation valves to the A/C units fail closed and are automatically closed on a Ventilation Isolation Actuation Signal (VIAS) to prevent CCW flow through the waterside economizers in a post-accident situation.

Technical Specification (TS) 2.12(1) requires that the temperature within the control room and control cabinets be maintained below 120°F does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in TS. However, the equipment required to maintain this temperature, the control room air conditioning system, meets Criterion 3 of 10 CFR 50.36 in that the system functions to mitigate a design basis accident by maintaining the control room in a habitable environment.

Therefore, it is proposed that this TS be revised to delete the control room temperature as a LCO and require that two control room air conditioning trains be operable when the reactor coolant temperature is above 210°F. The design temperature limits of instrumentation and controls inside of the control room will be maintained in the Basis Section of TS 2.12.

The allowed outage time for one train of control room air conditioning is proposed as 30 days. This is consistent with Combustion Engineering Standard TS 3.7.12 (NUREG-1432 Rev. 1). In addition, the FCS Probabilistic Risk Assessment model was reviewed and validated a 30 day outage time as being non-risk significant. The impact on Core Damage Frequency (CDF) from a 30 day LCO was based on the assumption that one cooling unit was always inoperable and thus under the LCO for an entire year. This allows the analysis to consider unlimited entries into the LCO and a full LCO duration for each entry. Using this assumption, the baseline (annually) CDF of 1.53E-5 would increase by 21.6% to a frequency of 1.86E-5. In accordance with EPRI's ''PSA Applications Guide,'' this small increase in CDF can be classified as ''non-risk significant.''

Specification 2.12(2)

Specification 2.12(2) requires that a thermometer be in the control room at all times. This instrumentation does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in the FCS TS. Therefore, the requirement is proposed for relocation to the FCS USAR.

Specification 2.12(3)

Specification 2.12(3) requires that all areas of the plant containing safety related instrumentation be observed during hot functional testing to determine local temperatures and monitored during operation if normal plant ventilation is not available. It is proposed to delete this TS. The requirement to monitor and determine local temperatures during hot functional testing was met during the initial startup phase of FCS and is no longer applicable. The requirement to monitor temperatures within the plant during normal operation does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in TS and therefore is being deleted.

The requirement to control temperatures for safety related instrumentation and controls, and initiate supplementary cooling if required, is currently described in USAR Section 9.10. These USAR requirements are controlled by plant procedures. Any changes to these requirements would require an evaluation be conducted in accordance with 10 CFR 50.59.

Specification 2.12(4)

Specification 2.12(4) allows one control room air filtration system to be inoperable for 7 days or a plant shutdown be commenced. This specification does not state which modes of operation it applies to.

Therefore, it is proposed to revise this specification to require two trains of control room air filtration systems to be operable when the reactor coolant temperature is above 210°F. The allowed outage time will be maintained at 7 days and a total of 42 hours will be allowed to take the plant to cold shutdown. The 42 hour time period is consistent with TS 2.0.1 which addresses equipment outages in excess of what is specifically allowed by individual specifications.

The proposed changes for the control room systems consist of providing additional restrictions on operation of the control room air filtration systems and control room air conditioning system. These changes ensure that equipment required to mitigate the consequences of an accident are operable. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

REFUELING OPERATIONS

The design bases of the fuel handling accident and refueling operations were reviewed and several inadequacies were identified related to refueling operations. Therefore, revisions are proposed for the TS Definition section, TS 2.6 on containment integrity, and TS 2.8 on refueling operations to reflect NUREG-1432.

Definitions

Cold Shutdown Condition & Refueling Shutdown Condition

The changes proposed for the definitions of Cold Shutdown Condition, and Refueling Shutdown Condition clarify these definitions. The plant is in Cold Shutdown when T_{cold} is less than 210°F, and the reactor coolant is at least Shutdown Boron Concentration but less than Refueling Boron Concentration. Similarly, the definition for Refueling Shutdown is clarified to apply when T_{cold} is less than 210°F and the reactor coolant is at least Refueling Boron Concentration. This change does not propose any new operating modes but merely clarifies when the definitions are applicable.

Core Alterations

The definition for Core Alterations is being revised to reflect the requirements of NUREG-1432. This revision deletes ''any component'' from the definition and clarifies that the components considered by this definition are those that could affect reactivity. In addition, the revision adds nuclear fuel to the definition such that movement of fuel within the reactor vessel will be defined as a core alteration and not a refueling operation.

Refueling Operations

The definition of Refueling Operations is being revised to delete control element assemblies (CEA) or startup sources from the definition since these are items that are included in the definition of Core Alterations. Additionally, it is being revised to specify that the definition is limited to movement of irradiated fuel outside of the reactor pressure vessel since fuel movement inside the reactor vessel is included in the definition of Core Alteration. Finally, a clarification is being added to state that suspension of refueling operations shall not preclude completion of movement of irradiated fuel to a safe, conservative position.

In Operation

The definition of In Operation is being revised to include the definition of operable. This is a more conservative interpretation than currently exists.

Specification 2.1 ''Reactor Coolant System''

It is proposed to revise TS 2.1.1(3) to include shutdown cooling requirements when the reactor coolant system (RCS) temperature is below 210°F with fuel in the reactor and the reactor vessel head fully tensioned. The definitions of Cold Shutdown (Mode 4) and Refueling Shutdown (Mode 5) contained in the TS make no distinction as to the status of the reactor vessel head or RCS temperature. The only difference between the two defined modes is boron concentration. Higher or lower boron concentration affects shutdown margin but does not affect decay heat load, which is the basis for this specification.

Technical Specification 2.1.1(4) was intended to address shutdown cooling requirements during refueling operations. However, this is already addressed in TS 2.8. Therefore, it is proposed to delete TS 2.1.1(4) and the exception since new specifications addressing shutdown cooling loop requirements during Mode 5 with fuel in the reactor and with one or more reactor vessel head closure bolts less than fully tensioned are proposed for inclusion in TS 2.8 (Refueling Operations).

The associated statements supporting these items in the Basis section are also proposed for deletion. Prior to any reactor vessel head closure bolts being loosened, TS 2.1.1 will be applicable which will require two shutdown cooling loops. As soon as a closure bolt is loosened, the new proposed TS 2.8 would be applicable which also requires two shutdown cooling loops whenever there is less than 23 feet of water above the core. The requirements of TS 2.1.1(3) are similar to NUREG-1432, Specifications 3.4.7 and 3.4.8.

Specification 2.6 ''Containment System''

Currently, TS 2.6(1)c states that containment integrity shall not be violated when the reactor vessel head is removed if the boron concentration is less than refueling concentration. However, Specification 2.6(1)c has no required actions and therefore, TS 2.0.1 must be entered when the LCO is not met. In this situation, (reactor vessel head removed), TS 2.0.1 is ineffective because the plant would already be in Refueling Shutdown. Thus, TS 2.6(1)c is proposed for deletion.

Currently, Specification 2.6(1)d requires that except for testing one control element drive mechanism at a time, positive reactivity changes shall not be made by CEA motion or boron dilution unless containment integrity is intact. Specification 2.6(1)d is proposed for deletion as it is unnecessarily restrictive.

Specification 2.8.1(1) as proposed eliminates the need for containment integrity when the reactor is in Refueling Shutdown. Specification 2.8.1(1) requires sufficient shutdown margin to preclude a criticality event and also prescribes actions to restore the shutdown margin if necessary. Small positive reactivity increases whether by CEA motion or boron dilution will not cause a criticality event due to the need to maintain at least a 5% shutdown margin. Therefore, the requirement to maintain containment integrity is unnecessarily restrictive since a criticality event cannot occur when a shutdown margin of at least 5% exists. Specification 2.8.1(1) is consistent with the requirements of NUREG-1432, Specification 3.9.1.

A new specification (TS 2.8.2(1)) is proposed that provides requirements for containment closure during core alterations and refueling operations inside of containment. The design basis of the Fort Calhoun Station does not require full containment integrity during a fuel handling accident. As stated in USAR Section 14.18, the fuel handling accident does not take credit for containment isolation. Therefore, requiring full containment integrity is inappropriate and requirements for containment closure are proposed for addition to TS 2.8 consistent with NUREG-1432 Specification 3.9.2.

Specification 2.10.2 governs operation of CEAs and monitoring of selected core parameters. Specification 2.10.2 ensures (1) adequate shutdown margin following a reactor trip, (2) that the moderator temperature coefficient (MTC) is within the limits of the safety analysis, and (3) CEA operation is within the limits of the setpoint and safety analysis. Specification 2.10.2 ensures that the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality and provides actions (i.e., boration) to be taken to ensure that the required shutdown margin is available. Thus, TS 2.10.2 precludes the need for containment integrity when the plant is in cold shutdown.

Specification 2.8 ''Refueling Operations''

It is proposed that TS 2.8 be rewritten to reflect NUREG-1432. Currently, this specification applies to any refueling operation. However, no distinction is made between refueling operations within containment and refueling operations within the spent fuel pool. In addition, several initial assumptions of a fuel handling accident are not addressed by the current TS 2.8.

Specification 2.8(1)

The current TS 2.8(1) is inadequate. This specification requires that the equipment hatch and one door in the Personnel Air Lock be properly closed, and all automatic containment isolation valves be operable or at least one valve closed. The specification does not define what is meant by a properly closed equipment hatch; that information is currently contained in the Basis of TS 2.1.1. In addition, inclusion of all automatic containment isolation valves instead of those providing direct access to the outside atmosphere is incorrect.

The containment isolation system is defined in USAR Section 5.9.5 as those devices actuated by a Containment Isolation Actuation Signal (CIAS) or a Steam Generator Isolation Signal (SGIS). This includes many valves that have no design basis function during a fuel handling accident. A CIAS is initiated by a Containment Pressure High Signal or a Pressurizer Pressure Low Signal. Neither of these signals are required to be operable during refueling operations as these signals would/could not respond to a fuel handling accident.

The correct requirements are specified in TS 2.8(2) which only requires that closure be initiated by the Ventilation Isolation Actuation Signal (VIAS) for the containment pressure relief, air sample, and purge system valves. Due to these inadequacies, it is proposed to delete TS 2.8(1) and replace it with a new Specification 2.8.2(1) which is consistent with NUREG-1432 Specification 3.9.3.

Specification 2.8(2)

It is proposed that TS 2.8(2) be deleted and replaced by new Specifications 2.8.2(3) and 2.8.3(5). The requirement to maintain an operable Ventilation Isolation Actuation Signal with input from the containment atmosphere gaseous and auxiliary building exhaust stack gaseous radiation monitors is consistent with current requirements and required actions are consistent with NUREG-1432, Specification 3.3.8. Radiation Monitor RM-052 functions as a ''swing'' monitor, i.e., it can be aligned to monitor either containment or the auxiliary building exhaust ventilation stack. Radiation Monitor RM-052 is powered by either MCC-3B1/AI-40C (like RM-051) or MCC-4C2/AI-40D (like RM-062).

Technical Specification 2.7, Electrical System is not required to be applied when the RCS is below 300°F. Above 300°F, TS 2.7 requires both 4160-VAC buses to be operable. Thus, above 300°F the required radiation monitors must be powered from independent 480-VAC buses supplied by independent 4160-VAC buses. During refueling outages, bus alignments other than those used during power operation are used to permit electrical system maintenance and modifications.

In the loss of offsite power event, the radiation monitor sample pumps and control room HVAC units stop and will not restart until the emergency diesel generators (EDGs) reenergize the system. The fuel handling equipment also stops and does not restart when the EDGs reenergize the system, thus minimizing the potential of a fuel handling accident. When the EDGs reenergize the buses, VIAS will operate as designed. Therefore, when the RCS is below 300°F, the required monitors need only be powered from independent 480-VAC buses supplied by a single 4160-VAC bus.

There is no need to assume that a fuel handling accident occurs immediately followed by a loss of offsite power. However, in the unlikely event that this should occur, there would be no effect on the site boundary dose since VIAS is not credited in USAR Section 14.18 (Fuel Handling Accident). In this situation, when the EDGs reenergize the buses, the control room HVAC units will restart in the filtered air makeup mode and the stack radiation monitor sample pump will restart. However, the containment radiation monitor sample lines remain isolated preventing the restart of the monitor sample pump after receipt of a VIAS.

Specification 2.8(3)

It is proposed that TS 2.8(3) be deleted. This requirement does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in the TS. The requirement that radiation levels in containment and the spent fuel pool shall be monitored during refueling operations will be incorporated into the FCS USAR.

Specification 2.8(6)

It is proposed that TS 2.8(6) be deleted. This requirement does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in the TS. The requirements that direct communication between personnel in the control room and at the refueling machine shall be available whenever core alterations are taking place will be incorporated into the FCS USAR.

Specification 2.8(7)

It is proposed that TS 2.8(7) be deleted and replaced with a new Specification 2.8.3(4). The requirement to place the spent fuel pool ventilation system in operation prior to refueling operations is consistent with the current TS. It is being clarified that this specification only applies to refueling operations in the spent fuel pool, and not when conducting refueling operations inside of containment. Additionally, it is being clarified that TS 2.0.1 is not applicable to this activity, as reactor operation is independent of fuel movements in the spent fuel pool.

Specification 2.8(9)

The current Specification 2.8(9) is inadequate. This specification requires a minimum of 23 feet of water above the top of the core. This does not meet the initial conditions assumed in the fuel handling accident as documented in USAR Section 14.18. USAR Section 14.18 assumes 23 feet of water above where the fuel could land if dropped. In order to meet this initial condition, a minimum of 23 feet of water above the reactor vessel flange is required, as this is the highest point where a fuel bundle could land if dropped. Procedures reflect the requirement to maintain 23 feet of water above the reactor vessel flange during refueling operations. The proposed revision is consistent with NUREG-1432, Specification 3.7.16.

Specification 2.8(11)

The current specification is inadequate. The specification provides restrictions on storage of fuel in the spent fuel pool; however, there are no required actions to address situations when the specification is not met. It is proposed that TS 2.8(11) be deleted and replaced with a new Specification 2.8.3(1) that requires that a misloaded fuel assembly be moved immediately. Additionally, it is being clarified that TS 2.0.1 is not applicable to this activity, as reactor operation is independent of fuel movements in the spent fuel pool.

Specification 2.8(12)

It is proposed that TS 2.8(12) be deleted and replaced with a new Specification 2.8.3(3). The requirement to maintain 500 ppm boron concentration in the spent fuel pool whenever unirradiated fuel is stored there is consistent with the current TS and the required actions are consistent with NUREG-1432, Specification 3.7.17.

Restriction on Movement of Irradiated Fuel from the Reactor Core

The restriction on irradiated fuel movement unless the core has been subcritical for at least 72 hours if the reactor has been operated at power levels above 2% is proposed for relocation to the Bases of TS 2.8.2(2). This requirement does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in the TS. This is consistent with NUREG-1432, B 3.9.6.

Reactor Coolant System Boron Concentration

Currently, there is no specification for boron concentration. Refueling boron concentration is included in the definition of Mode 5. However, there are no required actions to be taken if the boron concentration should be below refueling concentration. Therefore, it is proposed that a new Specification 2.8.1(1) be incorporated consistent with NUREG-1432, Specification 3.9.1.

Spent Fuel Pool Water Level

Currently, there is no specification for spent fuel pool water level. The water level of the spent fuel pool is an initial condition assumed in USAR Section 14.18. It is proposed that a new Specification 2.8.3(2) be incorporated into TS 2.8, which is consistent with NUREG-1432, Specification 3.7.16.

The proposed changes for the RCS and containment during shutdown, and requirements for refueling operations, consist of providing additional restrictions on operation, and changes to make the requirements of the TS Limiting Conditions for Operation consistent with the initial conditions and assumptions of the fuel handling accident as documented in USAR Section 14.18. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

SURVEILLANCE REQUIREMENTS

CONTROL ROOM

Specification 3.1, Table 3-3, Item 13.

Specification 3.1, Table 3-3, Item 13 requires that the thermometer in the control room be compared with a calibrated thermometer and replaced if out of tolerance on a refueling frequency. It is proposed that this surveillance be deleted to be consistent with deletion of the LCO requirement to maintain a thermometer in the control room.

A new surveillance is proposed to verify that the control room air conditioning system has the capability to remove the assumed heat load. This surveillance will ensure the operability requirements for TS 2.12 are met. The test and frequency is consistent with NUREG-1432.

The air-operated CCW isolation valves to the A/C units fail closed and are automatically closed on a VIAS to prevent CCW flow through the waterside economizers in a post-accident situation. These valves are currently tested in accordance with TS 3.3 (FCS Inservice Testing Program). Prior to the modification, the valves were tested as fail-open valves. No TS changes are necessary.

The control room air filtration system is currently tested on a refueling frequency in accordance with TS 3.2, Table 3-5, Item 10a. No TS changes are necessary.

REFUELING OPERATIONS

Reactor Coolant Boron Concentration During Refueling Operations

The Reactor Coolant System boron concentration is currently sampled in accordance with TS 3.2, Table 3-4, Item 1(e). It is proposed to revise the frequency from once per shift during refueling operations to once per 3 days which is consistent with NUREG-1432. As stated in the basis of TS 2.8 and USAR Section 14.18, the reactor cavity is filled with over 200,000 gallons of borated water prior to the start of refueling operations. The requirements for sampling the reactor coolant during the remainder of Mode 5 is performed once per 3 days in accordance with Table 3-4, Item 1(d). This proposed change will make the sampling consistent with the requirements of Item 1(d) and NUREG-1432.

Spent Fuel Pool Boron Concentration

The spent fuel pool boron concentration is currently sampled in accordance with TS 3.2, Table 3-4, Item 5. It is proposed to revise the frequency of the sampling to prior to movement of unirradiated fuel in the spent fuel pool and once per week whenever unirradiated fuel is stored there to be consistent with the requirements of the LCO.

Source Range Neutron Monitors

Currently, a channel check and calibration of the wide range neutron monitors is performed in accordance with TS 3.1, Table 3-1, Item 2.

Containment Penetrations

Currently, there is no surveillance to determine the status of containment penetrations during refueling operations. Therefore, a new surveillance is proposed for TS 3.2, Table 3-5 to verify the status of required containment penetrations once per 7 days consistent with NUREG-1432.

The requirement of NUREG-1432 to verify that the containment purge and exhaust valves actuate to the isolation position on a refueling frequency is currently tested as part of the Containment Radiation High Signal test required by TS 3.1, Table 3-2. Item 4.

Shutdown Cooling Loops

Currently, there is no surveillance requirement to verify that the required shutdown cooling loops are operable and in operation or to verify correct breaker lineup for the shutdown cooling loop that is not in operation. Therefore a new surveillance is proposed to be incorporated into TS 3.2, Table 3-5 consistent with NUREG-1432.

Refueling Water Level

Currently, there is no surveillance requirement to verify the refueling water level during refueling operations. Therefore, a new surveillance is proposed for incorporation into TS 3.2, Table 3-5 consistent with NUREG-1432.

Spent Fuel Pool Water Level

Currently, there is no surveillance requirement to verify the spent fuel pool water level during refueling operations. Therefore, a new surveillance is proposed for incorporation into TS 3.2, Table 3-5 consistent with NUREG-1432.

Spent Fuel Initial Enrichment/Burnup Verification

Currently, the requirement to conduct a verification of initial enrichment and burnup of spent fuel that will be stored in Region 2 is included as a general requirement of TS 2.8. It is proposed to relocate this requirement into a surveillance in TS 3.2, Table 3-5, consistent with NUREG-1432.

The proposed changes for the surveillance requirements consist of providing additional testing requirements to ensure that the Limiting Condition for Operations will be met. One surveillance frequency related to the sampling of the reactor coolant system boron concentration during refueling operations is being reduced from a frequency of once per shift to once every 3 days. However, this frequency is consistent with the frequency of sampling during the remainder of Mode 5 when fuel is in the

reactor and is more than adequate due to the large volume (over 200,000 gallons) of borated water required during refueling operations. Therefore, the proposed changes do not increase the probability or consequences of an accident previously evaluated.

ADMINISTRATIVE CHANGES

The remainder of TS 2.8 requirements of refueling operations are proposed to be reformatted into individual TS LCOs. It is also proposed that sampling frequencies of items contained in TS 3.2, Table 3-4, (page 3-19), be revised to incorporate frequencies defined in TS 3.0.2. Therefore, frequencies stated as once per 31 days will be noted as ''M,'' and frequencies stated as once per 7 days will be noted as ''W.'' These proposed changes have no effect on 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.

There will be no physical alterations to the plant configuration. No changes in operating modes are proposed although minor changes to the definitions of Cold Shutdown Condition and Refueling Shutdown Condition are proposed for clarification purposes. The proposed changes incorporate additional restrictions on the operation and testing of equipment required to mitigate an accident and to ensure the initial conditions and assumptions of the design basis accidents are maintained and controlled by the Technical Specifications.

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

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

The proposed changes ensure that assumptions of the fuel handling accident are maintained by Technical Specification Limiting Condition for Operation and surveillance requirements. The assumptions of the fuel handling accident that may affect a margin of safety are not being changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Project Director: William H. Bateman

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: February 25, 1997

Description of amendment request: The proposed Technical Specifications (TS) changes would amend the Limerick Generating Station (LGS) Unit 1 and Unit 2 Facility Operating Licenses (FOLs), and Appendix B of the licenses (i.e., Environmental Protection Plan (EPP)), reflecting a corporate name change from Philadelphia Electric Company to PECO Energy Company. In addition, the application would make changes to the LGS Units 1 and 2, FOL, and Appendix A (i.e., TS) of the licenses, which would remove obsolete information and correct typographical errors.

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

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

The company name change and typographical corrections are editorial and will not alter the operation of equipment assumed to be an initiator of any analyzed event or transients previously evaluated. The license provisions were satisfactorily completed, and as such, have no effect on any previously evaluated accident scenario. The changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients, nor will they alter the operation of equipment important to safety previously evaluated.

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

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

The company name change and typographical corrections are editorial and will not involve any physical changes to the plant systems, structures, or components. The license provisions were satisfactorily completed, and as such, have no effect on any previously evaluated accident scenario. The proposed changes do not allow plant operation in any mode that is not already evaluated. The changes will not alter the operation of equipment important to safety previously evaluated.

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

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

The company name change and typographical corrections are editorial and will not affect the manner in which the facility is operated, or change equipment or features which affect the operational characteristics of the facility. There is no margin of safety as defined in the bases of any TS regarding the name of the company, or affected by the corrections or deletion of obsolete license provisions.

Therefore, these 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.

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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

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: March 24, 1997

Description of amendment request: The proposed Technical specifications (TS) changes would delete the Drywell and Suppression Chamber Purge System operational time limit, and add a surveillance requirement to ensure the purge system large supply and exhaust valves are closed as required.

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

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

This activity does not increase the probability of occurrence of an accident previously evaluated in the SAR [Safety Analysis Report]. This activity involves deleting the allowable operating limit (180 hours each 365 days) for the Drywell and Suppression Chamber Purge system, while maintaining specific criteria for when the valves are allowed to be open. These changes do not increase the probability that this system will be in service should a LOCA [loss-of-coolant-accident] occur and does not increase the probability that a LOCA will occur. These changes also do not impact the probability of occurrence of any anticipated operational occurrence, other postulated design basis accident, or other event in which the plant was designed to respond.

This activity does not increase the consequences of an accident previously evaluated in the SAR. UFSAR [Updated Final Safety Analysis Report] Section 9.4.5.1.2.2 for high volume purging, although limiting the operating time the vent and purge system is to be in service, evaluates the consequences of a LOCA should the vent and purge valves be open. System operating procedures for venting and purging assure the availability of SGTS [standby gas treatment system] should a LOCA occur.

This activity will not increase the probability of a LOCA occurring during the time the Drywell and Suppression Chamber Purge system is in operation as previously evaluated. The Improved TS do not identify a specific time limit value as long as the valves are operated under the stated conditions (inerting, de inerting, pressure control, ALARA [as low as reasonably achievable] or air quality considerations for personnel entry or Surveillances that require that the valves be open). These proposed changes will incorporate the ITS [Improved Technical Specifications] operational controls which will result in the same order of magnitude of equipment malfunction probability as that provided by limiting purging to 180 hours per 365 days. A LGS [Limerick Generating Station] Level 2 PSA [Probability Risk Assessment] Analysis was performed to determine the additional risk associated with changing the operating limit from 90 hours to a nominal 500 hours each 365 days. This analysis concluded that the increase in risk of containment failure is well within the bounds of the EPRI [Electric Power Research Institute] PSA Applications Guideline for permanent changes and the NRC [Nuclear Regulatory Commission] Staff's safety goal value of 1.0 E-6 per year of reactor operation. Industry and LGS historical operating experience confirms that the purging lines are opened only for the specified reasons stated in ITS and for periods which do not exceed the current magnitude of equipment malfunction probability. Therefore, earlier engineering judgment is being replaced by operating experience.

Failure of the operating SGTS filter bank following a LOCA has been found to be acceptable due to the limited benefit derived from SGTS for accident sequences important to plant risk and the possibility that the backup filter bank would be available. Additionally, as discussed in UFSAR Section 9.4.5.1.2.2, the failure of SGTS during a LOCA does not contribute to any significant releases and is bounded by the analysis performed to address containment overpressure rupture.

Deleting the time limit restriction that the vent and purge line isolation valves may be open does not increase the probability that these valves will not perform as designed (close upon isolation signal) in response to a LOCA. Removing the 180 hour requirement will not increase the likelihood that the vent and purge valves will be called upon to close from that previously evaluated. UFSAR Section 6.2 states that the containment purge valves have undergone extensive testing and analyses to demonstrate the operability of these valves following a LOCA.

These changes do not directly or indirectly degrade the performance of any other safety system (assumed to function in the accident analysis) design basis. The potential for other equipment failures in the reactor enclosure due to duct impact, impingement, and the resulting environmental conditions was previously evaluated in the LGS SAR. It was concluded that the environmental qualifications for the LGS equipment are sufficient to ensure operability under the predicted environmental condition, and, the potential does not exist for impact or impingement - related damage to essential equipment. Maintaining the existing SAR analysis and retaining operating criteria for opening the containment purge valves, demonstrates that the risk of equipment failure and resulting radiological consequences will not increase.

Therefore, deleting the TS operating limit for the Drywell and Suppression Chamber Purge system from 180 hours each 365 days and the addition of a TS Surveillance Requirement verifying that the purge valves are closed under certain conditions does not increase the probability or consequences of an accident previously evaluated.

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

This activity does not change the function of the Drywell and Suppression Chamber Purge system, the containment isolation system, or SGTS as previously evaluated. Deleting the operational time limit that the vent and purge system is in service and the addition of a surveillance requirement does not create an accident initiator not already considered.

In addition, this activity does not create a failure mode not considered. All evaluated equipment failures that could occur as a result of a LOCA during high volume purging have previously been identified and evaluated. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The bases of TS 3.6.1.8 state that the 180 hour each 365 day operating limit for the Drywell and Suppression Chamber Purge system is imposed to protect the integrity of the SGTS filters. The LGS Offsite Dose Calculation Manual assures the availability of the backup SGTS filter train during operation of the vent and purge system. Furthermore, deleting the operating limit (180 hours each 365 days) does not reduce the margin of safety since specific criteria for opening the purge valves is being maintained and does not involve an increase in risk. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: 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

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: April 9, 1997

Description of amendment request: The proposed Technical Specifications (TS) changes would clarify existing battery specific gravity requirements, delete the requirement to correct specific gravity values based on electrolyte level, and allow the use of charging current measurements to verify the battery—s state of charge.

Basis for proposed no significant hazards consideration determination:

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

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

Changes to Technical Specifications surveillance requirements for specific gravity and Technical Specifications Bases commitments do not change the frequency or consequences of any accident previously evaluated. The proposed changes which commit to IEEE Standard 450-1995 for specific gravity testing, providing battery charging current as an alternate method to specific gravity measurements, and eliminating the commitment to perform electrolyte level correction do not prevent the DC system from performing its intended safety function. The proposed changes to the Technical Specification battery surveillance requirements and commitment to IEEE Standard 450-1995 for specific gravity are in accordance with current industry practices. These changes do not reduce the readiness and performance of the 1E DC power system to perform its intended function during a design basis event.

The proposed changes do not affect seismic specifications, separation criteria or environmental qualifications. The proposed changes do not impose an increase in or more severe test requirements, an increase in the frequency of operation, reduce independence or redundancy, modify the system or equipment protective features, introduce new equipment failures or impose additional loads than any previously evaluated. The Class 1E battery system will continue to meet all of the design standards applicable to the system and will not cause the system to operate outside of its design or testing limits.

Batteries or battery chargers and their failure are not initiators of the accidents previously evaluated. The proposed changes do not affect, degrade or prevent the response of active or passive systems described or assumed in the LGS accidents previously evaluated. In addition, the proposed TS changes will improve the availability of the station batteries.

Therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specifications changes which will revise the surveillance requirements and the TS Bases, do not increase the failure rate of the battery. The proposed changes clarify and enhance Operation's focus on the key battery parameters which will improve the availability of the station batteries. The station batteries are not accident initiators. The single failure of an electrical component was previously evaluated in the LGS accident analysis. Unexpected failures beyond the postulated single failure are no more likely to occur under the clarified surveillance requirements.

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

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

The revision clarifies and reduces the battery surveillance requirements for specific gravity. The revision eliminates the possibility for misinterpretation and provides consistency of the surveillance requirements. The specific gravity value for each connected cell is being revised to reflect a discrete number which meets the existing manufacturer's recommendations and does not differ from the value described in the present bases. LGS is currently committed to earlier revisions of IEEE Standard 450 (i.e., 1975 and 1980), and the incorporation of IEEE Standard 450-1995 for specific gravity will reflect current industry practices regarding specific gravity.

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

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

Local Public Document Room location: 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

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

Date of amendment request: April 30, 1997 (TS 97-01)

Description of amendment request: The proposed amendment would change the design features section of the Technical Specifications to provide for insertion of Lead Test Assemblies (LTAs) containing Tritium Producing Burnable Absorber Rods (TPBARs) in the Watts Bar Nuclear Plant (WBN) reactor core during Cycle 2. The purpose of the change is to provide irradiation services to support U.S. Department of Energy (DOE) investigations into the feasibility of using commercial light water reactors to maintain the DOE inventory of tritium.

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

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

LTAs do not adversely affect reactor neutronic or thermal-hydraulic performance; therefore, they do not significantly increase the probability of accidents or equipment malfunctions while in the reactor. The neutronic behavior of the LTAs mimics that of standard burnable absorbers with only slight differences which are accommodated in the core design. The reload safety analysis performed for WBN Unit 1, Cycle 2 will confirm that any minor effects of LTAs on the reload core will be within established fuel design limits.

As described in DOE Technical Report PNNL-11419, Revision 1, the LTA design is robust to all accident conditions except the large loss of coolant accident where the rods are susceptible to failure. However, the failure of the small number of TPBARS rods has been determined to have an insignificant effect on the thermal hydraulic response of the core to this event.

The impacts of LTAs on the radiological consequences for certain postulated events [as shown in Table 6-1 of the licensee's submittal, including Large Break LOCAs☐ are very small, and they remain within 10 CFR 100 regulatory limits. The additional offsite doses due to tritium leakage from the containment are small with respect to loss of coolant accident source terms and are well within regulatory limits.

The LTAs will not result in an increase in combustible gas released to the containment. Therefore, the LTAs do not result in a significant increase in the consequences of those previously considered.

Analysis has shown that TPBARs will not fail during Condition I through IV events, with the exception of a Large Break LOCA. The radiological consequences of the non-Large-Break LOCA events are essentially unchanged by the expected TPBAR tritium leakage to reactor coolant, and doses remain within a small fraction of 10 CFR 100 regulatory limits. Therefore, there is no significant increase in the consequences of these previously evaluated accidents.

The expected occupational and offsite doses, as reported in Technical report PNNL-11419, Revision 1, resulting from release of tritium from TPBARs over the plant operating cycle, including refueling, are not significantly increased and are within applicable regulatory limits.

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

LTAs have been designed to be compatible with existing Westinghouse 17x17 fuel assemblies and conventional Burnable Poison Rod Assembly (BPRA) handling tools, equipment, and procedures, and therefore no new accidents or equipment malfunctions are created by the handling of LTAs.

LTAs use materials with known and predictable performance characteristics and are compatible with PWR coolant. The LTA design has specifically included material similar to those used in standard burnable absorber rods with the exception of internal assemblies used in the production and retention of tritium. As described in the technical report, these materials are compatible with the reactor coolant system and the core design. For the irradiation proposed, the quantities of these materials is small. Therefore, no new accidents or equipment malfunctions are created by the presence of the LTAs in the reactor coolant system.

Thermal-hydraulic criteria have been established to ensure that TPBARs will not fail during Condition I or II events. Analysis has shown that TPBARs, appropriately positioned in the core, operate within the established thermal-hydraulic criteria. Therefore, no new accidents or equipment malfunctions are created by the presence of the LTAs in the reactor.

Analysis has shown that TPBARs will not fail during Condition III and IV events, with the exception of a large-break loss-of-coolant-accident. The radiological consequences of these events are small, with doses that are a small fraction of the 10 CFR 100 limits. Therefore there is no significant increase in consequences of these previously evaluated accidents.

LTAs do not adversely affect reactor neutronic or thermal-hydraulic performance; therefore, they do not create the possibility of accidents or equipment malfunctions of a different type than previously evaluated while in the reactor.

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

LTAs do not adversely affect reactor neutronic or thermal-hydraulic performance. Analysis indicates that reactor core behavior and offsite doses remain relatively unchanged. TPBAR performance under Condition I, II, III, and IV events are very similar to standard burnable absorber rods previously evaluated. For these reasons, the proposed amendment 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: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: April 18, 1997

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.3.2, ''Safety System Instrumentation,'' and TS Section 3/4.5.2, ''Emergency Core Cooling Systems - ECCS Subsystems - Tavg (greater than or equal to) 280°F.'' Certain surveillance intervals would be changed from 18 months to once each refueling interval, and certain setpoints would be changed. The associated bases would also be changed.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because the initiation of such accidents are not affected by the proposed revisions to increase the surveillance test intervals from 18 to 24 months for TS 3/4.3.2.1, ''Safety Features Actuation System Instrumentation,'' and TS 3/4.5.2, ''Emergency Core Cooling Systems - ECCS Subsystems - Tavg (greater than or equal to) 280—F.'' Initiating conditions and assumptions remain as previously analyzed for accidents in the DBNPS Updated Safety Analysis Report.

Results of the instrument drift study analysis and review of historical 18-month surveillance data and applicable maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or a new analysis has been performed incorporating a more conservative setpoint; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and applicable maintenance records reviews.

These proposed revisions are consistent with the NRC guidance on evaluating and proposing such revisions as provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' dated April 2, 1991.

The proposed revisions to Allowable Values for Safety Features Actuation System (SFAS) Reactor Coolant System (RCS) Pressure - Low, RCS Pressure - Low-Low, RCS Pressure - Low-Low bypass permissive, and Decay Heat Isolation Valve and Pressurizer Heater Interlocks have no bearing on the probability of the initiation of an accident previously evaluated.

The application of the Allowable Value to only the Channel Functional Test and not the Channel Calibration, the proposed deletion of the Trip Setpoints, the proposed revision of the TS 3.3.2.1 Limiting Condition for Operation (LCO) and Action Statement 3.3.2.1.a, and the proposed revisions to Actions 13 and 14 of TS Table 3.3-3, are associated with the proposed revision of the Allowable Values for SFAS RCS Pressure - Low, RCS Pressure - Low-Low, and Decay Heat Isolation Valve and Pressurizer Heater Interlocks, and are consistent with NUREG-1430, Revision 1, ''Standard Technical Specifications, Babcock and Wilcox Plants,'' dated April 1995. The proposed revisions have no bearing on the probability of the initiation of an accident previously evaluated.

The proposed changes to TS Bases 3/4.3.1 and 3/4.3.2, ''Reactor Protection System and Safety System Instrumentation,'' and TS Bases 3/4.5.2 and 3/4.5.3, ''ECCS Subsystems,'' are administrative changes associated with the other proposed changes, and do not affect previously analyzed accidents.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the slight increase in doses due to a letdown line break event as a result of the proposed change to the SFAS RCS Pressure - Low Allowable Value still satisfy the NRC Standard Review Plan Section 15.6.2 acceptance criteria that doses do not exceed a small fraction (10%) of the 10 CFR 100 guideline values. The remaining proposed changes to Allowable Values, and the other changes proposed by this License Amendment Request do not increase the radiological consequences of previously analyzed accidents because the source term, containment isolation, or radiological releases are not being changed by the proposed revisions.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated, for the reasons discussed below.

No changes are being proposed to the type of testing currently being performed, only to the length of the surveillance test interval.

Results of the instrument drift study analysis and review of historical 18-month surveillance data and maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or a new analysis has been performed incorporating a more conservative setpoint; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and applicable maintenance records reviews.

The proposed revisions to Allowable Values for SFAS RCS Pressure - Low, RCS Pressure - Low-Low, RCS Pressure Low-Low bypass permissive, and Decay Heat Isolation Valve and Pressurizer Heater Interlocks, do not alter the type of any testing currently being performed.

The application of the Allowable Value to only the Channel Functional Test and not the Channel Calibration, the proposed deletion of the Trip Setpoints, revision of the TS 3.3.2.1 LCO and Action Statement 3.3.2.1.a, and the proposed revisions to Actions 13 and 14 of

TS Table 3.3-3, are associated with the proposed revision to the Allowable Values for SFAS RCS Pressure - Low, RCS Pressure - Low-Low, RCS Pressure Low-Low bypass permissive, and Decay Heat Isolation Valve and Pressurizer Heater Interlocks, and are consistent with NUREG-1430, Revision 1, ''Standard Technical Specifications, Babcock and Wilcox Plants,'' dated April 1995. The proposed revisions do not alter the type of testing currently being performed.

The proposed changes to TS Bases 3/4.3.1 and 3/4.3.2, ''Reactor Protection System and Safety System Instrumentation,'' and TS Bases 3/4.5.2 and 3/4.5.3, ''ECCS Subsystems,'' are administrative changes associated with the other proposed changes, and do not alter any testing currently being performed.

3. Not involve a significant reduction in a margin of safety. The results of the instrument drift study analysis and review of historical 18-month surveillance data and applicable maintenance records support an increase in the surveillance test intervals from 18 to 24 months (and up to 30 months on a non-routine basis) because: the projected instrument errors caused by drift are bounded by the existing setpoint analysis or a new analysis has been performed incorporating a more conservative setpoint; and no potential for a significant increase in a failure rate of a system or component was identified during surveillance data and applicable maintenance records reviews. Existing system and component redundancy is not affected by these proposed changes.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences, consequently there are no significant reductions in a margin of safety.

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

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

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: April 18, 1997

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3/4.7.6, ''Plant Systems - Control Room Emergency Ventilation System.'' Additional Limiting Conditions for Operation (LCO) would be added related to the availability of the station vent normal range radiation monitoring instrumentation. The associated TS bases would also be modified consistent with these changes.

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

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

1a. Involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions, or assumptions are affected by the proposed changes.

The proposed change to LCO 3.7.6.1 would include new required Action statements in the event that one or both channels of station vent normal range radiation monitoring instrumentation become inoperable. In the event that one channel is inoperable for greater than 7 days, or in the event that both channels are inoperable, the proposed Action statement would require that the control room normal ventilation system be isolated and at least one Control Room Emergency Ventilation System (CREVS) train be placed in operation.

Under the proposed actions, the ventilation systems would be placed in a state equivalent to that which occurs were a high radiation isolation to occur. These proposed changes have no bearing on the probability of an accident.

The proposed change to the terminology utilized in Surveillance Requirement (SR) 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes have no bearing on the probability of an accident.

1b. Involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not change the source term, containment isolation, or allowable releases.

As described above, under the proposed new LCO 3.7.6.1 Actions, in the event that one station vent normal range radiation monitoring instrumentation channel is inoperable for greater than 7 days, or in the event that both channels are inoperable, the ventilation systems would be placed in a state equivalent to that which occurs were a high radiation isolation to occur. Therefore, in the unlikely event of an accident requiring control room isolation while in this condition, the dose consequences to control room operators would be unchanged.

The proposed change to the terminology utilized in SR 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes have no bearing on the consequences of an accident.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes.

As described above, under the proposed new LCO 3.7.6.1 Actions, in the event that one station vent normal range radiation monitoring instrumentation channel is inoperable for greater than 7 days, or in the event that both channels are inoperable, theventilation systems would be placed in a state equivalent to that which occurs were a high radiation isolation to occur. Operation of the equipment and components in this manner would not introduce the possibility of any new or different kinds of accidents.

The proposed change to the terminology utilized in SR 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes would not introduce the possibility of any new or different kinds of accidents.

3. Involve a significant reduction in a margin of safety because the proposed changes to the Action under LCO 3.7.6.1 ensure that control room isolation capability is maintained in the event a station vent radiation monitor is inoperable. The proposed allowable outage time of 7 days for one inoperable channel is consistent with the presently allowable outage time for one inoperable CREVS. The proposed Action to place at least one CREVS train in operation within 1 hour, in the event both channels of radiation monitoring become inoperable, is more conservative than the present Action which requires that a plant shutdown commence within 1 hour, but does not require the CREVS be placed in operation.

The proposed change to the terminology utilized in SR 4.7.6.1.e is an administrative change made to make the terminology consistent with the proposed new Action statements. The proposed changes to Bases 3/4.7.6 are administrative changes consistent with the proposed changes to LCO 3.7.6.1. These changes would not affect the margin of safety.The NRC staff has reviewed the licensees' analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Basis for proposed no significant hazards consideration determination:

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: August 22, 1996

Description of amendment request: The proposed change would remove the action statement of Technical Specification (TS) Section 3.2.G, Table 3.2.6, Note 7, requiring reactor shutdown after 30 days of inoperability of the high range stack gas monitor and substitute an action statement consistent with the guidance provided in NRC Generic Letter 83-36.

The high range stack monitor provides an estimate of gross stack activity that has exceeded the upper limit of the normal range instrumentation. The high range monitor reading serves as input to dose projection systems for initial estimation of off-site conditions. The monitor reading would be used prior to the acquisition of stack isotopic sample data which would provide a more accurate indication of stack activity.

The licensee stated, among other things, that due to the passivefunction of the instruments and the ability to monitor this parameter utilizing alternate methods, it is not appropriate to impose stringentrequirements on the operation of the unit. This monitor is identified in the Vermont Yankee Regulatory Guide 1.97 submittal as Category 2, Type E. This monitor provides post-accident information for use in determining the magnitude of the release of radioactive materials and for monitoring such release. However, the high range stack monitor does not have any safety function associated with the prevention or automatic mitigation of design-basis accidents, neither does it provide primary information needed to permit the control room operating personnel to take required manually controlled actions.

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

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

The High Range Stack Monitor is a RG [Regulatory Guide] 1.97, Category 2, Type E instrument with no specified safety function associated with the prevention or automatic mitigation of design basis accidents, neither does it provide primary information needed to permit the control room operating personnel to take required manually controlled actions. The proposed change to the action statement associated with this monitor will not change the function of this monitor, and since the monitor is not assumed to initiate any accidents, nor function to mitigate any accidents, this change will not significantly increase the probability or consequences of any previously analyzed accident.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure effective methods are available to assess post accident conditions. Thus, this 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.]

The proposed change to the action statement associated with this monitor will not change the function of this monitor, and since the monitor is not assumed to function for the prevention or mitigation of any previously evaluated accidents, this change will not significantly reduce a margin of safety.

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

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

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Patrick D. Milano, Acting

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

Date of amendment request: January 24, 1997, as supplemented on May 15, 1997 (TSCR 193)

Description of amendment request: The proposed amendments (Point Beach Nuclear Plant (PBNP) Technical Specifications (TS) Change Request (TSCR) 193) would revise TS 15.5.4, ''Fuel Storage,'' to increase fuel assembly enrichment limits to 5.0 w/o U-235 while maintaining K_{eff} in the storage pools (spent fuel pool and new fuel storage racks) less than 0.95. The May 15, 1997, supplement provided a revised no significant hazards consideration determination that superseded the licensee's determination noticed in the **Federal Register** on April 23, 1997 (62 FR 19837).

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

1. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not involve a change to structures, systems, or components that would affect the probability or consequences of an accident previously evaluated in the PBNP Final Safety Analysis Report (FSAR). The only relevant concern with respect to increasing enrichment limits in the spent fuel pool and new fuel storage racks is one of criticality. The proposed changes use the same criticality limit used in the current Technical Specifications. Therefore, margin to safe operation of Units 1 and 2 is maintained. The probability and consequences of an accident previously evaluated are dependent on this criticality limit. Because the limit will not change, the probability and consequences of those accidents previously evaluated will not change.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a change to the physical structure of the spent fuel pool or of the plant. The proposed increase in spent fuel pool and new fuel storage racks fuel assembly enrichment limits maintains the margin to safe operation of Units 1 and 2 because the criticality limit for the spent fuel pool and new fuel storage racks will not change. The enrichment increase does not affect any of the parameters or conditions that contribute to the initiation of any accidents. Because the criticality limit remains the same, these changes have no effect on plant operation or on the initiation of any accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The proposed changes maintain the margin to safe operation of Units 1 and 2. The margin of safety is based on the criticality limit of the spent fuel pool and the new fuel storage racks. Because this limit will not change, the margin of safety will not be affected. Therefore, the proposed changes will not create 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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: John N. Hannon

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

Date of amendment request: April 23, 1997

Description of amendment request: This request proposes to revise Technical Specification 3/4.9.4, Containment Building Penetrations, and its associated Bases section, to allow selected containment isolation valves to be opened under administrative controls during periods of core alterations or movement of irradiated fuel inside containment.

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

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

The proposed change involves changes to the Technical Specification requirements for containment closure which is an accident mitigating feature. The changes would not affect the likelihood of occurrence of any accidents previously evaluated. The proposed change does not involve any hardware or plant design changes. The containment leakage value is not assumed to be an initiator of any analyzed event. Containment isolation valves and temporary closure devices serve to limit the radiological consequences of accidents. The proposed change would ensure the service air and breathing air manual isolation valves will perform their required containment closure function and will serve to limit the consequences of a fuel handling accident as described in the USAR, such that the results of the analyses in the USAR remain bounding. In considering the consequences of a design basis fuel handling accident inside containment, the assumptions in the analysis take no credit for the containment as a barrier to prevent the postulated release of radioactivity. For events that could occur during CORE ALTERATIONS or movement of irradiated fuel assemblies, containment closure is considered a defense-in-depth boundary to prevent uncontrolled release of radioactivity. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change involves reliance on manual actuation of containment penetration valves (Service Air valves KA V-039 and KA V-118 and Breathing Air valves KB V-001 and KB V-002 are manual valves) to block the unimpeded flow of the containment atmosphere to the environment under certain conditions. The proposed change would not necessitate a physical alteration of the plant features that provide core cooling or subcriticality (no new or different type of equipment will be installed) or changes in parameters governing plant operation during CORE ALTERATIONS or movement of irradiated fuel in containment. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change is similar to the use of administrative controls to isolate an open containment airlock door. The use of administrative controls in this manner has been approved by the NRC (WCGS Technical Specification Amendment 95) for plant operations that would not require the containment to maintain a pressure boundary. This scenario is applicable during plant shutdown for refueling when CORE ALTERATIONS and movement of irradiated fuel assemblies in the containment occur. Accidental damage to spent fuel during these operations is classified as a fuel handling accident. The proposed change has been developed considering the importance of the containment boundary in limiting the consequences of a design basis fuel handling accident. The proposed change allows for protection equivalent to that provided by previously approved methods of containment closure. Considering the probability of an event that would challenge the containment boundary, the alternative protection provided by this change, and the operational requirements to occasionally open these penetrations, the proposed change is acceptable and any reduction in the margin of safety is insignificant.

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

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

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

NRC Project Director: William H. Bateman

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: January 24, 1997, as supplemented March 27, 1997

Brief description of amendment: The proposed amendment will update the

Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification 2.1.2 and the associated Bases section to reflect the results of the latest cycle-specific calculation performed for the Pilgrim Nuclear Power Station Operating Cycle 12. In addition, the values provided in Note 5 of Table 3.2.C.1, which are based on the SLMCPR values, have been revised as a result of the changes to the SLMCPR value.

Date of issuance: April 7, 1997

Effective date: April 7, 1997

Amendment No.: 171

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

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6568) The March 27, 1997, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 7, 1997 No significant hazards consideration comments received: No

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

Commonwealth Edison Company, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: August 19, 1996, as supplemented on February 5, March 13, April 29 and April 30, 1997.

Brief description of amendments: The amendment would revise Technical Specification (TS) Section 4.4.5.2 to extend, for one additional operating cycle (i.e., Cycle 7), the 1.0 volt and 3.0 volt interim plugging criteria (IPC) which were added to the Braidwood, Unit 1, TSs by License Amendment No. 69, issued on November 9, 1995. Additionally, this amendment to the Braidwood, Unit 1, license added some definitions and reporting requirements to TS Section 4.4.5.2 and modified the designations for the IPC models in TS Bases Section 3/4.4.4.5. Braidwood, Unit 1, Cycle 7, will end in fall 1998. While there are no revisions to the TS for Braidwood, Unit 2, both units are being amended to maintain the continuity of the amendment numbers.

Date of issuance: May 14, 1997.

Date of effective: Immediately, to be implemented within 30 days.

Amendment Nos.: 82

Facility Operating License Nos. NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 12, 1997 (62 FR 6570). The February 5, March 13, April 29 and April 30, 1997, submittals provided clarifying technical information that did not affect the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 14, 1997. No significant hazards consideration comments received: No

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

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: June 20, 1996, as supplemented December 30, 1996, and March 5, 1997.

Brief description of amendments: The amendments would change the TSs by incorporating an NRC-approved thermal limit licensing methodology in the list of approved methodologies used in establishing the fuel cycle-specific thermal limits. In addition, the proposed amendments would change the TSs to reflect the use of Siemens Power Corporation (SPC) ATRIUM-9B fuel for the first time at Dresden, Units 2 or 3. The proposed amendments would also correct minor editorial items in the TSs.

In March 1997, the NRC staff performed an audit of the application of Advanced Nuclear Fuel for Boiling Water Reactors (ANFB) to ATRIUM-9 fuel. The staff raised concerns associated with the ATRIUM-9B fuel additive constant uncertainty used as input to the NRC-approved methodology for the calculation of minimum critical power ratio (MCPR). In response to the audit findings, by letter dated April 18, 1997, SPC submitted a generic topical report (ANF-1125(P) Supplement 1 Appendix D), which is currently under staff review, for the future reload analysis in the safety limit MCPR calculation. The staff schedule for the review of the topical report will not be timely enough for the resolution of the ATRIUM-9B MCPR issue to support reload and restart of Dresden, Unit 3. Therefore, by letters dated May 2 and May 6, 1997, ComEd provided additional information concerning the MCPR issues and how it will affect the Dresden, Unit 3, D3R15 fuel cycle and provided additional information concerning the ATRIUM-9B fuel design and shutdown margin that are applicable during refueling and shutdown.

The staff is currently reviewing the licensee's May 2 and May 6, 1997, letters. To be more timely and support the reload schedule for Dresden, Unit 3 (currently scheduled for May 20, 1997), the staff has chosen to split its consideration of the proposed amendments into two parts. The first part of the amendment package now being evaluated would modify Section 5.3.A, ''Design Features'' of the TSs to reflect use of the ATRIUM-9B fuel design and would include two SPC topical reports in TS Section 6.9.A.6, ''Core Operating Limits Report,'' to reflect mechanical design criteria for this fuel. This change would allow this fuel to be loaded into the core only under Operational Modes 3 (Hot Shutdown), 4 (Cold Shutdown), and 5 (Refueling) and does not permit startup or power operation using the ATRIUM-9B fuel.

Date of issuance: May 16, 1997

Date of effective: Immediately, to be implemented within 30 days.

Amendment Nos.: 159 and 154

Facility Operating License Nos. DPR-19 and DPR-25: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17227). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 16, 1997 No significant hazards consideration comments received: No

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Commonwealth Edison Company, Docket No. 50-265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois

Date of application for amendment: April 21, 1997

Brief description of amendment: The amendment increases the minimum critical power ratio safety limit for Unit 2 and adds a Siemens Power Corporation reference to the Technical Specifications (TS) to allow plant operation in Operational Modes 1 and 2.

Date of issuance: May 22, 1997

Date of effective: Immediately, to be implemented within 30 days.

Amendment No.: 174

Facility Operating License No. DPR-30: The amendment revised the TSs. Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 23499 dated April 30, 1997). This notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by May 30, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final no significant hazards consideration determination are contained in a Safety Evaluation dated May 22, 1997.

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

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan Date of application for amendment: December 2, 1996 (NRC-96-0134)

Brief description of amendment: The amendment revises TS 3.1.4.3, TS Table 3.3.6-1, and TS Table 4.3.6-1 to change the operability requirements for the Rod Block Monitor (RBM). Specifically, the revision requires the RBM to be operable when reactor thermal power is greater than or equal to 30 percent of rated thermal power.

Date of issuance: May 15, 1997

Date of effective: May 15, 1997, with full implementation within 60 days

Amendment No.: 112

Facility Operating License No. NPF-43. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** January 2, 1997 (62 FR 124) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1997. No significant hazards consideration comments received: No.

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

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

Date of application of amendments: April 29, 1997

Brief description of amendments: The amendments incorporate a license condition that will allow revisions to the Oconee Updated Final Safety Analysis Report (UFSAR) that clarifies the main turbine-generated missile protection criteria.

Date of issuance: May 16, 1997

Date of effective: As of the date of issuance and implementation is the incorporation in the UFSAR the changes described in Duke Power Company's application dated April 29, 1997

Amendment Nos.: 224, 224, and 221

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the UFSAR and added a new License Condition. Public comments requested as to proposed no significant hazards consideration: Yes. (62 FR 24512 dated May 5, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received as of the date of issuance. The notice also provided for an opportunity to request a hearing by June 9, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 16, 1997.

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

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

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: November 26, 1996, as supplemented February 12, 1997.

Brief description of amendment: The amendment changes the allowable primary-to-secondary leak rate and in the Surveillance Requirements section of the TSs it changes the acceptance criteria for steam generator tubes. The amendment changes the reference that is included in the tube acceptance criteria from Combustion Engineering topical report CEN-601-P Revision 01-P to CEN-630-P, Revision 01.

Date of issuance: May 20, 1997

Date of effective: May 20, 1997, to be implemented within 30 days.

Amendment No.: 184

Facility Operating License No. NPF-6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 4, 1996 (61 FR 64376) The February 12, 1997, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: April 4, 1995, as supplemented by letters dated August 25, 1995, and April 18, 1997.

Brief description of amendment: The amendment changes the required frequency for inspecting reactor coolant pump flywheels.

Date of issuance: May 20, 1997

Date of effective: May 20, 1997, to be implemented within 30 days.

Amendment No.: 185

Facility Operating License No. NPF-6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 5, 1995, (60 FR 35069) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 1997. No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One,Unit No. 2, Pope County, Arkansas

Date of application for amendment: October 7, 1996, as supplemented February 10, and May 8, 1997

Brief description of amendment: The amendment changes the channel functional testing frequency for most of the Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation from monthly to every four months. In addition, the amendment allows the use of Cycle Independent Shape Annealing Matrix (CISAM) methodology in the Core Protection Calculators (CPCs). Finally, the amendment makes a number of administrative changes to the Technical Specifications (TS) to clarify the existing TS or correct previous errors in the TS.

Date of issuance: May 21, 1997

Date of effective: May 21, 1997

Amendment No.: 186

Facility Operating License No. NPF-6: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4346) The Commission's related evaluation of the amendment is contained in a Safety

Evaluation dated May 21, 1997 No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

Date of amendment request: March 27, 1997

Brief description of amendment: The amendment changes TSs surveillance requirements 4.5.2.d.3 and 4.5.2.d.4 by increasing the required amount of trisodium phosphate dodecahydrate (TSP) stored in the containment sump from 97.5 cubic feet to 380 cubic feet, and adjusts the TSP sampling requirement accordingly.

Date of issuance: May 15, 1997

Date of effective: May 15, 1997, to be implemented within 60 days.

Amendment No.: 127

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

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17234) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 1997 No significant hazards consideration comments received: No.

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

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

Date of amendment request: August 21, 1996, as supplemented by letter dated March 17, 1997

Brief description of amendment: The amendment approves revision of Attachment 1 to the operating license concerning design and testing modifications in the Containment Vacuum Relief System (CVR) that penetrate the primary containment at Waterford Steam Electric Station, Unit 3. The penetrations affected are commonly referred to as Penetrations 53 and 65.

Date of issuance: May 20, 1997

Date of effective: May 20, 1997, to be implemented within 90 days.

Amendment No.: 128

Facility Operating License No. NPF-38: Amendment revised the Operating License.

Date of initial notice in **Federal Register:** November 6, 1996 (61 FR 57484) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 1997. No significant hazards consideration comments received: No.

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

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: September 13, 1996, as supplemented by letter dated January 15, 1997.

Brief description of amendment: The amendment revised the Technical Specifications to permit the use of 10 CFR Part 50, Appendix J, Option B, performance-based containment leakage rate testing.

Date of issuance: May 19, 1997

Date of effective: May 19, 1997, to be implemented within 60 days of the date of issuance.

Amendment No.: 158

Facility Operating License No. DPR-36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 6, 1996 (61 FR 57487) The January 15, 1997, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 19, 1997. No significant hazards consideration comments received: Yes. Comments were submitted by Patrick J. Dostie on behalf of the State of Maine by letter dated April 15, 1997. The staff responded by letter dated May 19, 1997.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

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

Date of amendment request: February 18, 1997, as supplemented by letter dated February 26, 1997.

Description of amendment request: The amendment revises the Appendix A Technical Specifications relating to the reactor core fuel assembly design features requirements contained in Technical Specification 5.3.1, Fuel Assemblies. The changes made by this amendment allow for the limited replacement of failed or damaged fuel rods in fuel assemblies with solid stainless steel or zirconium alloy filler rods.

Date of issuance: May 13, 1997

Date of effective: May 13, 1997

Amendment No.: 51

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 12, 1997 (62 FR 11496) The licensee's letter dated February 26, 1997, provided a correction to a typographical error in the original application but does not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 13, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 2, 1995.

Brief description of amendment: This amendment changes the TS to reflect changes in the organization as they apply to oversite and management of the Trojan Nuclear Plant.

Date of issuance: October 31, 1996

Date of effective: October 31, 1996

Amendment No.: 195

Facility Operating License No. NPF-1: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 27, 1995 (60 FR 58404) No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 7, 1997

Brief description of amendments: These amendments revise Technical Specification (TS) 3/4.2.5 to incorporate an exception to the provisions of TS 4.0.4 and to clarify the time at which the surveillance can be performed by adding that the surveillance is to be performed within 24 hours after attaining steady state conditions at or above 90% rated thermal power. The revised surveillance contains editorial enhancements that clarify the surveillance requirement. Salem Unit 1 TS Table 3.2-1 is also being revised to delete reference to three loop operation.

Date of issuance: May 8, 1997

Date of effective: Both units, as of date of issuance, to be implemented prior to entry into Mode 1 from the current outage. Amendment Nos. 193 and 176

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

Date of initial notice in **Federal Register:** January 29, 1997 (62 FR 4353) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 8, 1997. No significant hazards consideration comments received: No

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

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

Date of amendment request: March 25, 1997

Brief description of amendments: The amendment changes 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 40 percent from 0.5 [micro]Ci/gram to 0.3 [micro]Ci/gram and the maximum instantaneous value would be reduced by 40 percent from 30 [micro]Ci/gram to 18 [micro]Ci/gram.

Date of issuance: May 19, 1997

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

Amendment Nos.: 128

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

Date of initial notice in **Federal Register:** April 4, 1997 (62 FR 16201) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 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

Dated at Rockville, Maryland, this 28th day of May, 1997.

For the Nuclear Regulatory Commission

Jack W. Roe,

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

[Doc. 97–14395 Filed 6–3–97; 8:45 am]

BILLING CODE 7590–01–F

SECURITIES AND EXCHANGE COMMISSION

[Rel. No. IC–22686; 811–4068]

Pacifica Funds Trust; Notice of Application

May 28, 1997.

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

ACTION: Notice of Application for Deregistration under the Investment Company Act of 1940 (the ''Act'').

APPLICANT: Pacifica Funds Trust.

RELEVANT ACT SECTION: Section 8(f).

SUMMARY OF APPLICATION: Applicant seeks an order declaring that it has ceased to be an investment company.

FILING DATES: The application was filed on January 31, 1997, and amended on May 9, 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 June 23, 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 may request notification of a hearing by writing to the SEC's Secretary.

ADDRESSES: Secretary, SEC, 450 5th Street, N.W., Washington, D.C. 20549. Applicant, 237 Park Avenue, Suite 910, New York, NY 10017.

FOR FURTHER INFORMATION CONTACT: Deepak T. Pai, Staff Attorney, at (202) 942–0574, or H.R. Hallock, Jr., Special Counsel, 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.

Applicant's Representations

1. Applicant is an open-end management investment company organized as a Massachusetts business trust. On July 16, 1984, applicant registered under the Act and filed a registration statement on Form N–1A pursuant to section 8(b) of the Act. The registration statement became effective on November 30, 1984. Applicant commenced an initial public offering of the first of its 23 series on December 26, 1985, and commenced its last initial public offering of a series on November 15, 1995. Shares of five series were never offered to the public.

2. First Interstate Capital Management, Inc., served as applicant's investment adviser prior to April 1, 1996, when its parent company, First Interstate Bancorp, merged into Wells Fargo & Company. At a meeting on May 17, 1996, applicant's board of trustees, including a majority of the trustees who are not ''interested persons'' of applicant, approved entry into an Agreement and Plan of Reorganization (the ''Reorganization Agreement'') by and between applicant and Stagecoach Funds, Inc. (''Stagecoach''), an open-end investment company advised by Wells Fargo Bank, N.A. In reviewing the proposed reorganization, applicant's board considered the potential impact of the reorganization on applicant's shareholders, including (a) provisions intended to avoid the dilution of shareholder interests; (b) the capabilities, practices, and resources of the organizations that provided investment advisory and certain other services to applicant and Stagecoach; (c) the shareholder services provided to applicant's shareholders, compared with the shareholder services provided to Stagecoach shareholders; (d) the investment objectives, policies and limitations of each series of applicant and the corresponding series of Stagecoach; (e) the historical investment performance of each series of applicant and the corresponding series of Stagecoach; (f) the historical and projected operating expenses of each series of applicant and the corresponding series of Stagecoach; and (g) the anticipated tax consequences of the reorganization.

3. Based upon its evaluation of the information presented, applicant's board of trustees determined that the reorganization was in the best interests of the shareholders of each series of applicant, and that the interests of the shareholders of each series would not be diluted. An amendment to the Reorganization Agreement was subsequently approved by the applicant's board of trustees on August 15, 1996, which provided that, because of tax considerations, certain liabilities of one of applicant's 23 series (Pacifica Asset Preservation Fund) would be retained by that series rather than transferred to its corresponding series of Stagecoach.

4. On or about June 6, 1996, proxy materials for a special shareholders meeting were distributed to applicant's