NUCLEAR REGULATORY COMMISSION

[Docket No. 50–356]

Environmental Assessment and Finding of No Significant Impact; Regarding Issuance of a Specific Exemption to the Requirements of 10 CFR 50.82(b)(6)(ii); University of Illinois at Urbana-Champaign; Low Power Reactor Assembly

The U.S. Nuclear Regulatory Commission (NRC) is considering granting, for Facility Operating License No. R–117 for the University of Illinois at Urbana-Champaign (the licensee or University) Low Power Reactor Assembly (LOPRA) located on the licensee's campus in Urbana, Illinois, a specific exemption in accordance with 10 CFR 50.12 to the part of the requirements of 10 CFR 50.82(b)(6)(ii) that requires a terminal radiation survey and associated documentation to demonstrate that the facility and site are suitable for release as a condition of license termination.

Environmental Assessment

Identification of Proposed Action

By application dated February 10, 1995, as supplemented on April 24, 1995, and October 2, 1996, the licensee requested authorization to decommission the LOPRA in accordance with the proposed decommissioning plan, and terminate Facility Operating License No. R–117. Amendment No. 6 to the facility operating license was issued on January 21, 1997, approving the decommissioning plan. The licensee informed the NRC in a letter dated April 15, 1997, that the University has completed decommissioning of the LOPRA in accordance with the amendment. The NRC project manager for the LOPRA and a non-power reactor inspector visited the site on May 7, 1997, and found that the licensee had decommissioned the LOPRA in accordance with the license amendment and that no licensed material remained under the authority of the LOPRA license. The licensee had transferred the LOPRA components and fuel to the Advanced TRIGA Research Reactor (TRIGA) license (Docket No. 50–151, Facility License No. R–115). Some components containing byproduct material were subsequently transferred to a University of Illinois byproduct materials license (License IL–01271–01), issued by the State of Illinois to allow the components to be stored at a facility away from the Nuclear Reactor Laboratory.

The University's Nuclear Reactor Laboratory houses the TRIGA (which the University continues to operate) and housed the LOPRA, which was located in the bulk shielding tank of the TRIGA. The Nuclear Reactor Laboratory continues to be subject to the terms of the TRIGA license. The Nuclear Reactor Laboratory will be considered for release by NRC as part of the request to terminate the TRIGA license at some time in the future. Because the facility and site will continue to be used under an NRC license and will be surveyed in the future, and because application of the regulation is not necessary to achieve the underlying purpose of the rule, the licensee requested in its letter of April 15, 1997, that NRC consider granting a specific exemption in accordance with 10 CFR 50.12 to the part of the requirements of 10 CFR 50.82(b)(6)(ii) that requires a terminal radiation survey and associated documentation to demonstrate that the facility and site are suitable for release as a condition for license termination.

The Need for Proposed Action

The exemption is needed for termination of Facility Operating License No. R–117.

Environmental Impact of Granting of Exemption

No licensed material remains under the authority of the LOPRA license. The NRC staff has verified that the LOPRA components and fuel have been transferred to the TRIGA license and the University of Illinois byproduct materials license, issued by the State of Illinois, which are authorized to receive this material. Future use of these components and fuel as a subcritical assembly in the TRIGA bulk shielding tank is currently authorized by the TRIGA license. With the transfer of all licensed material from the LOPRA license, the termination of the LOPRA license is administrative in nature. Because the facility and site will continue to be used under an NRC license, and because no facility or site is to be released as part of the license termination, granting the exemption will have no effect on the status of the site and, thus, no significant impact on the environment.

Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denying the proposed action. Not granting the exemption would not change current environmental impacts and would require continuance of Facility Operating License No. R–117. The staff also considered taking no action. This would have the same outcome as not granting the proposed action. The environmental impacts of the proposed action and of the alternative actions are similar. Since the LOPRA components and fuel have been transferred to other licenses that are authorized to receive this material, there is no alternative with less environmental impact than granting the exemption, which would allow the termination of Facility Operating License No. R–117.

Agencies and Persons Consulted

The staff consulted with the Illinois State official regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

The NRC has determined not to prepare an environmental impact statement for the proposed action. On the basis of the foregoing environmental assessment, the NRC has concluded that the granting of the exemption will have no significant effect on the quality of the human environment.

For further details with respect to this proposed action, see the application for termination of Facility Operating License No. R–117, dated February 10, 1995, as supplemented, which includes the letter of April 15, 1997, which requests the exemption. These documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20037.

Dated at Rockville, Maryland, this 9th day of July 1997.

For the Nuclear Regulatory Commission.

Marvin M. Mendonca,

Acting Director, Non-Power Reactors and Decommissioning Project Directorate, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 97–18665 Filed 7–15–97; 8:45 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 June 23, 1997, through July 3, 1997. The last biweekly notice was published on July 2, 1997 (62 FR 35846).

Notice Of Consideration of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Harards Consideration Determination, And Opportunith 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 August 15, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: May 30, 1997, identified as CY-97-006

Description of amendment request: Changes to the Operating License, DPR-61, and facility Technical Specifications (TS) that reflect the permanently shut down and defueled status of the plant.

CY-97-006 contains the proposed changes to the license conditions in DPR-61 on Fire Protection, Power Level and Fuel Movement; and submittal of a new set of TS referred to by the licensee as the Defueled TS (DTS). The DTS contain a revised Definitions section, removal of the sections on Safety Limits and Limiting Safety System Settings, Limiting Conditions for Operation and Surveillance Requirements were modified extensively, the Design Features section was revised, and the Administrative Controls section was modified to reflect all the preceding 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:

Connecticut Yankee Atomic Power Company (CYAPCO) has reviewed the proposed changes to the Operating License and the Technical Specifications in accordance with 10 CFR 50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:1. Involve a significant increase in the probability of consequences of an accident previously evaluated.

Because of the present plant configuration, many of the postulated accidents previously evaluated (i.e., loss or coolant accident, main steam line break, etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased. In fact, there is more likely a decrease in probability of a fuel handling accident since the need to move/rearrange fuel assemblies is minimal until they are removed from the spent fuel pool (i.e., for dry cask storage or for transferring to U.S. Department of Energy possession).

The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident. With the plant defueled and permanently shutdown, the demands on the radwaste systems is lessened since no new radioisotopes are being generated by irradiation or fission. Therefore, there is no increase in the probability or consequences of a gaseous or liquid radioactive release.

The changes to the Operating License reflect the permanently defueled condition for power level and fuel movement restrictions and the fire protection regulation which is applicable for a permanently defueled plant.

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the Technical Requirements Manual is due to the reduced heat load in the spent fuel pool.

The plant was shutdown on July 22, 1996 and more than 280 days have passed since the shutdown, thus the heat load on the spent fuel pool cooling system is greatly reduced. Present cooling performance data as well as calculations demonstrate that either the plate or the shell and tube heat exchanger has more than adequate heat removal capacity. In the event of a loss of forced cooling, calculations indicate that the spent fuel pool time to boil is greater than 40 hours based on an initial pool temperature of 150°F. The initial pool temperature of 150°F is based on Technical Specification 3/4.9.15 which has a pool temperature limit of 150°F. Even during boiling, the fuel is adequately cooled. Once boiling commences, the operators have in excess of 18 days to provide forced cooling and/or makeup before there is inadequate shielding provided by the water in the pool. This allows sufficient time to provide for alternate forced cooling or makeup to the spent fuel pool in the event of a service water system failure. Therefore, operability of spent fuel pool cooling does not require service water, electrical power, or makeup water to be immediately available.

Should failure to restore operation of the spent fuel pool cooling system occur before boiling takes place, cooling of the spent fuel can be accomplished by allowing the spent fuel pool to boil and adding makeup water at a rate equal to or greater than the boil-off rate.

CYAPCO has in place procedures to establish onsite power in the event of a Loss of Normal Power (LNP) and in the event of a loss of cooling to the Spent Fuel Pool. For a LNP, power can be made available within approximately one hour. If onsite power cannot be reestablished, due to equipment failure, at approximately 2 hours into the LNP, limited makeup water could be provided by gravity feed from a tank (available in approximately 30 minutes) or an unlimited supply of water could be provided via the diesel fire pump from the Connecticut River (available in approximately 30 minutes). Therefore, within approximately 2 1/2 hours of the event start, cooling and/or makeup would be reestablished to the spent fuel pool. Historically, the longest LNP the HNP has experienced has been less than 30 minutes.

The changes to Technical Specification 3.3.3.8, ''Radioactive Gaseous Effluent Monitoring Instrumentation'' and Table 3.3.-10 delete the trip function from the main stack noble gas activity monitor. The changes to Technical Specifications 3.11.2.1, Dose Rate, and 3.11.2.3, Dose, delete the requirement to include the radioiodine isotopes in the dose calculations. These changes are based on the following:

There is no significant increase in the consequences of a fuel handling accident since the accident scenarios assume an assembly with significant amounts of radioactive iodine or noble gas. The plant was shutdown on July 22, 1996. Except for I-125 (half-life =59.5 days), I-129 (half-life = 1.6E7 years), and Kr-85 (half-life =10.8 years), the spent fuel inventory of the dose contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for ''Dose Equivalent I-131'' (≥Standard Technical Specifications, Westinghouse Plants,'' NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible inventory in the spent fuel. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary are a small fraction of the 1O CFR 100 dose limits and the EPA PAGS. In fact, due to this decreased radioactive inventory, there is a significant decrease in the consequences of a fuel handling accident.

Based on the above, the proposed changes to the Operating License and the Technical Specifications 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.

There is no change in how spent fuel is stored or moved in the spent fuel pool. Therefore, the postulated fuel handling accidents are still bounding and are still considered as credible postulated accidents. The bases provided in the CYAPCO analysis of previously evaluated accidents in Section 1, above, also applies to the possibility of new or different accidents herein.

Based on the analysis in Section 1, above, the changes to Technical Specification related to radioactive iodine and noble gas isotopes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Based on these considerations, the proposed changes to the Operating License and the Technical Specifications 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.

With respect to the Service Water System (Specification 3/4.7.3), Electrical Power Systems (Specification 3/4.8) and spent fuel pool makeup, the basis for placing appropriate requirements in the Technical Requirements Manual is due to the reduced heat load in the spent fuel pool.

The Technical Specification basis states that the time to spent fuel pool boiling after a loss of forced cooling following a full core offload is 7 hours.

In accordance with the analysis set forth above under No. 1, there is no change in how spent fuel is stored or moved in the spent fuel pool.

Based on the above, the proposed changes to the Operating License and the Technical Specifications 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: Russell Library, 123 Broad Street, Middletown, CT 06457

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270

NRC Project Director: Marvin M. Mendonca, Acting Director

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

Date of amendment request: May 30, 1997, identified as CY-97-024

Description of amendment request: CY-97-024 provided the proposed technical specifications (TS) needed to implement the Certified Fuel Handler (CFH) program at the plant. This new position will replace the former licensed operator positions. A copy of the CFH Training Program, ''Nuclear Training Manual NTM-7.083'' was enclosed with the license amendment request for NRC review and approval. However, this manual will be reviewed separately from the proposed TS changes and when the NRC review of the manual is completed a letter of approval will be sent to the licensee.

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:

Connecticut Yankee Atomic Power Company (CYAPCO) has reviewed the proposed changes to the Technical Specifications in accordance with 10 CFR 50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not.

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

The proposed qualification, staffing and training requirements are appropriate for the present plant conditions.

The plant has permanently ceased operations, the reactor has been permanently defueled, and the spent fuel stored in the spent fuel pool.

Because the present plant conditions, many of the postulated accidents previously evaluated (i.e., loss-of-coolant accident, main steam line break, etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased. In fact, there is more likely a decrease in probability of a fuel handling accident since the need to move/rearrange fuel assemblies is minimal until they are removed from the spent fuel pool (i.e., for dry cask storage or for transferring to U.S. Department of Energy possession).

There is no significant increase in the consequences of a fuel handling accident since the accident scenarios assume an assembly with significant amounts of radioactive iodine or noble gas. The plant was shutdown on July 22, 1996. Except for I-125 (half-life=59.5 days), I-129 (half-life=1.6E7 years), and Kr-85 (half-life-10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). In addition, the definition for ''Dose Equivalent I-131'' (≥Standard Technical Specifications, Westinghouse Plants,'' NUREG-1431) does not include I-125 and I-129 in the dose assessment due to their negligible spent fuel inventory. Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e., whole body and thyroid) at the Exclusion Area Boundary and the Low Population Zone are a small fraction of the 10 CFR 100 dose limits. In fact, due to this decreased radioactive inventory, there is a significant decrease in the consequences of a fuel handling accident.

The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident. With the plant defueled and permanently shutdown, the demands on the radwaste systems are lessened since no new radioisotopes are being generated by irradiation. Therefore, there is no increase in the consequences of a gaseous or liquid radioactive release.

Based on the above, the proposed changes to the Technical Specifications 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.

There is no change in how spent fuel is stored or moved in the spent fuel pool. Therefore, the postulated fuel handling accidents are still bounding and are still considered as credible postulated accidents.

Based on the above, the proposed changes to the Technical Specifications 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.

There is no change in how spent fuel is stored or moved in the spent fuel pool.

The plant was shutdown on July 22, 1996. Except for I-125 (half-life=59.5 days), I-129 (half-life=1.6E7 years), and Kr-85 (Half-life=10.8 years), the spent fuel inventory of the dose-contributing radioactive iodine and noble gas isotopes has decayed more than 20 half-lives since shutdown (i.e., less than 0.0001% of the original amount remains). Except for Kr-85, the other noble gas nuclides that contribute to a whole body dose have also decayed to a negligible amount. CYAPCO has performed fuel handling and cask drop accident dose calculations which conclude that doses (i.e, whole body and thyroid) at the Exclusion Area Boundary and the Low Population Zone are a small fraction of the 10 CFR 100 dose limits.

Therefore, there is no significant reduction the margin of safety. In fact, due to this decreased radioactive iodine inventory, there is more likely an increase in the margin of safety.

Based on the above, the proposed changes to the Technical Specifications 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: Russell Library, 123 Broad Street, Middletown, CT 06457

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270

NRC Project Director: Marvin M. Mendonca

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan

Date of amendment request: June 20, 1997 (NRC-97-0037), as supplemented by letter dated July 3, 1997

Description of amendment request: The proposed amendment would relocate technical specification surveillance requirement 4.4.1.1.2 for the reactor recirculation system motor-generator (MG) set scoop tube stop setpoints to the Updated Final Safety Analysis Report. In addition, the proposed amendment includes the following changes to the surveillance testing methodology: (1) eliminating any licensing basis requirement for the electrical stops, and (2) revising the periodicity from a calendar basis to a situational basis (i.e., plant conditions that would dictate a change in stop positions).

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

The proposed change removes from the Fermi 2 Technical Specifications (TS) a Surveillance Requirement (SR 4.4.1.1.2) that is an implementation detail and relocates it to the Updated Final Safety Analysis Report (UFSAR), where it is more adequately and more appropriately controlled in accordance with 10 CFR 50.59. In addition, this proposed change revises the test methodology by: (1) eliminating the requirement for the electrical stops because they are not credited for mitigating any transients or accidents, and (2) revising the periodicity from a calendar basis to a situational basis to coincide with the beginning of each operating cycle or post-maintenance. These changes do not eliminate the necessary testing of the MG set mechanical stops. The MG set mechanical stops will continue to remain operable because the recirculation pump MG set mechanical speed stop settings will continue to be maintained at or below the required limits. The $MCPR_f$ [minimum critical power ratio] and $MAPLHGR_f$ [maximum average planar linear heat-generation rate] limits, along with the recirculation pump MG set mechanical speed stop settings on which they are based, are specified in the Core Operating Limits Report and operation within these limits is required by Technical Specifications 3.2.1 and 3.2.3. The changes described will therefore have no impact on the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification change does not result in any changes to the design (equipment/configuration) or operation of the plant and will thus not create a new failure mode or common mode failure. The MG set mechanical stops will continue to operate as intended and as designed. These changes will therefore not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

Changes in the methodology and frequency of testing will not involve a significant reduction in the margin of safety because the testing necessary to ensure the stops are set correctly will continue to be performed. Additionally, the $MCPR_f$ and $MAPLHGR_f$ limits, along with the recirculation pump MG set mechanical speed stop setting that they are based on, are specified in the Core Operating Limits Report, and operation within these limits is still required by Technical Specifications 3.2.1 and 3.2.3. Therefore, the margin of safety as defined in the bases of any Technical Specification is not reduced by relocating the surveillance requirement from the TS to the UFSAR. In addition to the above, relocation of the TS is consistent with the BWR Improved Standard Technical Specification, NUREG-1433, Rev. 1.

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

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226

NRC Project Director: John N. Hannon

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

Date of amendment request: April 24, 1997

Description of amendment request: The requested amendment revises the inservice inspection requirements associated with steam generator tube sleeves.

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

1. Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

This change implements a more stringent surveillance requirement than currently exists. It incorporates a requirement to inspect a minimum of 20% of each type of installed sleeve in each steam generator. The 20% inspection criterion is conservative with respect to the existing requirement of a 3% initial inspection of all steam generator tubes. Additionally, since the process for inspections has not changed, the probability or consequences of accidents previously analyzed are not increased as a result of inspection activities. The proposed changes have no impact on any previously analyzed accident in the safety analysis report.

The administrative changes made to update the technical specifications or to correct inconsistencies introduced in previous amendments do not affect reactor operations or accidental analyses and have no radiological consequences.

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

2. Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The changes made to increase the initial sample of sleeved tubes inspected during a surveillance, to update the technical specifications and to correct inconsistencies introduced in previous amendments are administrative and do not change the design, configuration or method of operation of the plant nor does it introduce any new possibility for an accident.

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

3. Does Not Involve a Significant Reduction in the Margin of Safety.

As previously discussed, this change implements a more stringent surveillance requirement than currently exists. The existing technical specifications require an initial inspection of 3% of the tubes in each steam generator while the proposed change requires inspection of a minimum of 20% of each type of installed sleeve. The 20% inspection criterion is conservative with respect to the existing technical specification. Existing technical specification operability and surveillance requirements are not reduced by the proposed change, thus no margins of safety are reduced.

The other administrative changes do not reduce technical specification operability and surveillance requirements, and therefore, do not reduce any margin of safety.

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

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

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

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

NRC Project Director: William D. Beckner

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

Date of amendment request: June 26, 1997

Description of amendment request: The proposed amendment will modify Technical Specification (TS) Tables 3.7-1 and 3.7-2. Table 3.7-1 will be revised to change the Main Steam Safety Valves (MSSVs) orifice size from 26 square inches to 28.27 square inches and to relocate the orifice size from the TS Table to the TS Bases. The change to correct the orifice size is an editorial change to make the TS consistent with plant design. Table 3.7-2 will be revised by deleting the provision that allows continued plant operation with three MSSVs inoperable. The proposed amendment will also revise TS Bases 3/4.7.1.1 to remove the equation used for determining the reduced maximum allowable linear power level-high reactor trip settings of TS Table 3.7-2.

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

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

Response: No

In response to the ABB/CE report pursuant to 10CFR21 regarding the omission of Main Steam Safety Valve (MSSV) piping pressure loss in safety analyses, the proposed change will eliminate the ability to operate the plant in accordance with Technical Specification 3.7.1.1 Action a with three MSSVs inoperable. The Bases to this Technical Specification will also be revised to state that the acceptability for operation at lower power levels with one or two MSSVs inoperable will be determined from results obtained from a loss of condenser vacuum accident analysis under these conditions. Deleting the allowance for continued operation with three MSSVs inoperable does not increase the probability of an accident. The consequences of an accident will not be increased by these changes. These changes are more restrictive and ensure that the MSSVs maintain their safety function of removing adequate heat from the steam generator in order to maintain peak steam generator pressure and peak pressurizer pressure well below their respective acceptance criteria during normal operation and all anticipated operational occurrences.

Changing the MSSVs orifice size listed in TS to their actual size and the orifice size utilized in the safety analysis, and relocating the MSSVs orifice size to the Technical Specification Bases does not affect the probability or consequences of an accident. The correct orifice size was used in the safety analysis and it is not subject to change unless a station modification is performed which will require a 10CFR50.59 evaluation and revision of the safety analysis. The MSSVs orifice size can be adequately controlled in the TS Bases which will also require a 10CFR50.59 to be changed.

Therefore, operation of Waterford 3 in accordance with this proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change will eliminate the ability to operate the plant in compliance with Technical Specification 3.7.1.1 Action a with three MSSVs inoperable. The Bases for this Technical

Specification will also be revised to state that the acceptability for operation at lower power levels with one or two MSSVs inoperable will be determined from results obtained from a loss of condenser vacuum accident under these conditions. The proposed change also revises the MSSVs orifice size to reflect the actual orifice size and the orifice size utilized in the safety analysis, and relocates the orifice size from Technical Specifications to the Technical Specification Bases. The proposed change does not involve any new equipment, components, or modifications and does not create any new system interactions or connections. Therefore, operation of Waterford 3 in accordance with this proposed change will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

Response: No

The proposed change will ensure that all appropriate acceptance criteria for the MSSVs are met during normal operation and all anticipated operational occurrences. The Technical Specification Bases 3/4.7.1.1 will be updated to state that the acceptance criteria for operation in accordance with Technical Specification 3.7.1.1 Action a will be determined from the results of the limiting loss of condenser vacuum accident. This change ensures that the transient and dynamic effects which occur during accident scenarios are fully evaluated. These changes also ensure that the MSSVs will maintain peak steam generator pressure and peak pressurizer pressure well below their respective acceptance criteria during normal operation, design basis accidents and anticipated operational occurrences.

The proposed change also revises the MSSVs orifice size to reflect the actual orifice size and the orifice size utilized in the safety analysis, and relocates the orifice size from Technical Specifications to the Technical Specification Bases. This change corrects an editorial error in the Technical Specifications and relocates unsurveilled design details from the Technical Specifications. Adequate control of the orifice size will remain adequate because any changes to the orifice size or the orifice size listed in the Bases will require a station modification and a TS Bases change. Station Modifications and TS Bases changes requires evaluation in accordance with 10CFR50.59.

Therefore, operation of Waterford 3 in accordance with this proposed change will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: James W. Clifford, Acting

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: The proposed amendment to Technical Specifications 3.9.1.2 and 3.9.13 and their Bases would allow crediting soluble boron for maintaining k-effective at less than or equal to 0.95 within the spent fuel pool (SFP) rack matrix following a seismic event of a magnitude greater than or equal to an operating basis earthquake (OBE).

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.

There is one Spent Fuel Pool accident condition discussed in Chapter 15 of the FSAR [Final Safety Analysis Report]. The FSAR discusses a fuel handling accident which drops a fuel assembly onto the fuel racks during fuel movement. Degradation of the Boraflex panels in a post-seismic condition will have no effect on the probability of a fuel assembly drop onto the stored fuel, or the fuel racks. Changing the way Boraflex responds to a seismic event will have no impact on the probability of a seismic event. A misplaced fuel assembly can be postulated in the MP3 [Millstone Unit 3] fuel pool as a result of either equipment malfunction or operator error. Degradation of the Boraflex panels will have no effect on the probability of a fuel misplacement event. Therefore, the degradation of Boraflex in a post-seismic condition does not involve an increase in the probability of an accident previously evaluated.

A fuel handling accident could cause a radioactive release of fission gases, resulting in dose consequences. This radioactive release of fission gases is due to the failure of a certain number of fuel pins which are postulated to fail during the fuel handling accident. The number of fuel pins which are postulated to fail in this event is not affected by the degradation of the Boraflex panels in a post-seismic condition. There are no criticality issues with this fuel handling accident for the reasons described next. Should a fuel handling accident occur prior to a seismic event, the existing fuel handling accident/misloading criticality analysis is still valid, such that 800 ppm [parts per million] of soluble boron is sufficient to ensure that K-effective of the SFP is maintained at less than 0.95. Although overly conservative, should a fuel handling accident occur during or after a seismic event, even with no Boraflex credit, the proposed 1750 ppm of soluble boron is sufficient to ensure that K-effective of the SFP is maintained at less than 0.95. Therefore, this proposed change does not involve an increase in the probability or consequences 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 change in the way Boraflex in conjunction with the addition of 1750 ppm boron responds to a seismic event does not create a new accident. The use of soluble boron in the Spent Fuel Pool is safe during and immediately following a seismic event, because the balance of the equipment in the fuel building not connected to the fuel pool which could cause a dilution (firewater, hot water heating, and demineralized water, CCP [component cooling-plant]) are seismic or mounted in such a fashion as to not direct unborated water into the fuel pool should a line rupture. Non borated water sources that are connected to the SFP will be isolated following a seismic event of greater than or equal to [an] OBE to prevent dilution. Therefore there is no possibility of [an] SFP boron dilution accident coincident with a seismic event, and credit for soluble boron is acceptable to meet the K-effective limit of 0.95 for the SFP. The crediting of soluble boron in the Spent Fuel Pool to control K-effective following a seismic event does not create a new accident as boron dilution of the pool can be prevented by closing and administratively controlling the opening of dilution paths to the pool and initiating routine sampling requirements on SFP boron. At present the crediting of soluble boron following a fuel misplacement event is allowed for the Millstone 3 Spent Fuel Pool. Analysis has shown that a seismic event of greater than an OBE level earthquake can be more limiting than a fuel misplacement event. As such the minimum boron requirement in the fuel pool will be increased from 800 ppm to 1750 ppm. As such, no new accident has been created because the crediting of boron following a malfunction/accident has always been an allowed event.

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 margin of safety, as defined by MP3 Technical Specifications, is to ensure that the K-effective of the MP3 SFP is maintained less than or equal to 0.95 at all times. There is no reduction in the margin of safety as the result of the degradation of Boraflex following a greater than an OBE seismic event, because soluble boron can be used to compensate for the loss of Boraflex. A value of 1750 ppm of soluble boron in the SFP at all times ensures that K-effective of the MP3 SFP is maintained less than or equal to 0.95 at all times, including this new malfunction of degraded Boraflex following a greater than an OBE seismic event.

Eliminating the credit for the negative reactivity effect of Boraflex panels in conjunction with the addition of 1750 ppm boron will have no effect on the probability of a seismic event. As the probability of a seismic event has not changed there is no increase in the probability of an accident or malfunction due to a seismic event. Following a seismic event operators are presently required to make inspections of the plant to determine post seismic event plant conditions. As a result of this change, inspections will be required to post seismic event evaluations to review the status of the Spent Fuel Pool and isolate potential dilution paths. These action are consistent with present guidance in the seismic response procedure and do not create an undue burden on the operator. To compensate for the potential

loss of Boraflex after a seismic event, the SFP is now required to be borated at all times to 1750 ppm to maintain the proper post seismic [K-effective] condition. As such there is no mitigation equipment that has to operate in the Spent Fuel Pool following a seismic event.

Although the Boraflex in the fuel racks is assumed to fail in a greater than an OBE seismic event, the presence of soluble boron in the fuel pool water will compensate for the loss of Boraflex. Surveillance requirements on SFP boron will ensure that there will be boron present in the SFP and ensure that the SFP is not diluted below the minimum required boron concentration during normal operation.

As the presence of SFP soluble boron during and after a seismic event maintains [K-effective] less than 0.95 there is no effect on the consequences of any malfunctions evaluated. As there are no new accidents created and there are no changes in the probability or consequences of previously analyzed accidents there is no effect on the consequences of any accident. There is no reduction in the margin of safety as the result of the degradation of Boraflex following a greater than an OBE seismic event, because soluble boron can be used to compensate for the loss of Boraflex to maintain K-effective less than 0.95.

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

In conclusion, bases 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: March 26, 1997

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to incorporate additional restrictions on the operation of the main steam safety valves (MSSVs).

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 Omaha Public Power District (OPPD) proposes to revise the Fort Calhoun Station (FCS) Unit No. 1 Technical Specifications (TS) 2.1.6, ''Pressurizer and Main Steam Safety Valves,'' to incorporate additional restrictions on the Main Steam Safety Valves (MSSVs) as a result of recent engineering analyses.

FCS has two Steam Generators (SG), each with one 2 1/2-inch MSSV and four 6-inch MSSVs. The purpose of the MSSVs is to limit the secondary system pressure to less than or equal to 110% of the design pressure of 1000 lbs. per square inch absolute (psia) when passing 100% of design steam flow.

The pressure drops in the main steam lines were calculated. The total losses (line losses and valve losses) of 30.5 psid (2 1/2 inch valves) and 33.5 psid (6 inch valves) were compared to the valve blowdown which is adjusted/checked each refueling outage as part of the required surveillance test. The pressure losses are less than the 39 psid and 40 psid blowdown for the 2 1/2 inch and 6 inch valve with the lowest setpoint (respectively). Therefore, the recommendation from the Part 21 to review blowdown settings to preclude valve chatter was conducted and there is no concern at FCS. A review of existing calculations for line losses in the primary system was conducted and was determined to be 39 psid for the inlets to the primary safety valves.

Analyses were then conducted to determine the impact of the total line losses on previously analyzed accidents documented in the Updated Safety Analysis Report (USAR). The scope of the analyses was to evaluate the pressure drops in the piping run for both the primary and MSSVs to determine the impact on the peak primary and secondary system pressures. The applicable transient for peak primary system pressure is the Loss of Load, and for maximum secondary system pressure is the Loss of Feedwater. All analyses were performed using the NRC-approved CESEC-III transient analysis methodology and computer code.

The assumptions of the analyses were that the plant is operating at 1535.6 MWt, (100% power + 2% uncertainty + reactor coolant pump heat), the MSSVs lifted at +3% of their nominal setpoints, the primary safety valve setpoints were adjusted to account for line losses and lifting at +1% of their setpoints, and the pressure losses in the main steam line to the SG were added to obtain the maximum secondary system pressure within the SG. Additional cases were evaluated with a +6% primary safety valve drift since this possibility is described in the Bases to TS 2.1.6.

The results from these analyses confirm that the effective increase in MSSV set pressure caused by the piping pressure losses leading to the primary safeties and MSSVs is below the 1100 psia design limit for the secondary system, and below the 2750 psia design limit for the primary system. This is predicated on the fact that only one (1) MSSV may be inoperable per SG.

Failure of a MSSV is not an initiator of any previously analyzed accident, and therefore the proposed changes do not increase the probability of an accident previously analyzed. The proposed change to revise TS 2.1.6 to allow only one MSSV per SG to be inoperable has been shown, utilizing NRC approved methodology, to limit the design pressure to values below the design limits. An administrative change to revise the TS setpoint value for both the primary safety valves and MSSVs from pounds absolute to pounds gauge is proposed to be consistent with the nameplate values of the valves and has no effect on any analyses. Therefore the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There will be no physical alterations to the plant configuration, changes in operating modes, setpoints, or testing methods. The additional restrictions being incorporated into the TS on MSSV operation will ensure that the design basis limits of 110% of design pressure will be met for the primary and secondary systems for analyzed accidents when considering inlet pipe pressure drops. The possibility of valve chatter being caused by the additional pressure losses identified in the Main Steam lines and MSSVs was reviewed and is not a concern. This is due to the valve blowdown (the difference between a valve's opening pressure and closing pressure) being greater than the pressure losses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change results in a peak primary pressure of 2649 psia (with 1% primary safety valve drift as allowed by TS 2.1.6) and peak secondary pressure of 1081 psia for the loss of load event compared to 2632 psia and 1075 psia documented in USAR Section 14.9. The proposed change results in a peak primary pressure of 2562 psia and peak secondary pressure of 1090 psia for the loss of feedwater event compared to 2487 psia and 1052 psia documented in USAR Section 14.10. The analyses confirm that the primary and secondary systems will continue to be below their respective design limits of 2750 psia and 1100 psia. 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: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

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

NRC Project Director: William H. Bateman

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: May 31, 1996

Description of amendment request: This change deletes Technical Specification 4.7.2.d.2, ''Control Room Emergency Outside Air Supply System Surveillance Requirement,'' related to the detection of chlorine.

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

Review of the various design basis accidents identified in Chapter 15 of the Susquehanna SES [Steam Electric Station] Final Safety Analyses Report (FSAR) concluded that none of these accidents are affected by deletion of the chlorine detection surveillance requirement from Technical Specifications. With the elimination of bulk quantities of gaseous chlorine from use at Susquehanna SES the probability of control room inhabitability due to a gaseous chlorine release has actually decreased. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change involves only the deletion of the chlorine detection system Technical Specifications based upon a plant modification to remove gaseous chlorine as a biocide from Susquehanna SES and replace it with an oxidizing biocide with non-gaseous/non-volatile properties. The release of chlorine from an off-site source is bounded by Reg. [Regulatory] Guide 1.95 in that manual isolation capability for the control room ventilation system is acceptable. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would not alter the margins of safety provided in the existing FSAR analysis (Sections 2.2.3.1.3 and 6.4) for chlorine release events since the basis for the existing margin of safety, which are the Reg. Guide 1.95 requirements, are not altered by the change. As stated above, since gaseous chlorine is no longer used for open cooling water treatment at Susquehanna SES and since the biocide currently used does not pose the same personnel inhalation threat as gaseous chlorine, safety margin has actually increased. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: John F. Stolz

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

Date of amendments request: June 13, 1997

Description of amendments request: The proposed amendments would change Technical Specification (TS) 3/4.9.13, ''Storage Pool Ventilation (Fuel Movement),'' by adding a note in the TSs to specifically indicate that the normal emergency power source may be inoperable in MODE 5 or 6 provided that the requirements of TS 3.8.1.2 are satisfied and extend the TS 3.9.13 completion time allowed for returning one out-of-service penetration room filtration system from 48 hours to 7 days. The Bases will also be modified to provide additional detail concerning these 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:

(1) The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. The proposed changes have no impact on the probability of an accident. The storage pool ventilation system will continue to ensure that radioactive material released as a result of a fuel handling accident in the spent fuel pool room will be filtered through the HEPA [high efficiency particulate air] filters and charcoal absorbers prior to discharge to the atmosphere. There is no change in the FNP [Farley Nuclear Plant] design basis as a result of this change and, as a result, does not involve a significant increase in the consequences of an accident previously evaluated.

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

(3) The proposed changes do not involve a significant reduction in the margin of safety. As a result of these proposed changes, the penetration room filtration system, when it is aligned to the spent fuel pool room, will continue to require verification of operability. There is no impact in the accident analyses. These proposed changes are technically consistent with the requirements of NUREG-1431, Revision 1 which has already received the requisite review and approval of the NRC staff. Thus the proposed changes do not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of amendment request: April 29, 1997, as supplemented by letter dated May 28, 1997

Description of amendment request: The amendment would revise the Unit 1 reactor vessel pressure and temperature limits to reflect data collected from the material sample recovered during the March 1996 Unit 1 outage.

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

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

Pressure and Temperature (P/T) limits for the reactor pressure vessel are established to the requirements of 10 CFR [Part] 50, Appendix G to ensure brittle fracture of the vessel does not occur.

This revision changes the P/T curves in the Unit 1 Technical Specifications to reflect the material capsule surveillance results from the sample removed during the [s]pring outage of 1996.

The RPV [reactor pressure vessel] surveillance capsule contained flux wires for neutron flux monitoring and Charpy V notch impact and tensile test specimens. The irradiated material properties were compared to available unirradiated properties to determine the effect of irradiation on material toughness for the base and weld materials through Charpy testing. Irradiated tensile testing results are compared with unirradiated data to determine the effect of irradiation on the stress-strain relationship of the materials.

The P/T curves are modified to reflect the results of the above examination. These curves and their operating limits were evaluated using the approved methodologies of 10 CFR [Part] 50 Appendix G and ASME [American Society of Mechanical Engineers] Code Appendix G. The new curves therefore represent the latest information available on the state of the reactor vessel materials. The P/T curves are generated for reactor vessel protection against brittle fracture, they do not affect the recirculation piping. Accordingly, the probability of occurrence of a design basis Loss of Coolant Accident (LOCA) is not increased. Likewise, no other previously evaluated accident and transients, as defined in Chapter 14 of the Final Safety Analysis Report (FSAR) are affected by this proposed change to the Unit 1 P/T curves. Additionally, this proposed revision does not affect the design, operation, or maintenance of any safety related system designed for the mitigation or prevention of previously analyzed events.

Since no previously evaluated accidents or transients are being affected by this change, their probability of occurrence is not increased and their consequences are not made worse.

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

Implementing the proposed P/T curves into the Unit 1 Technical Specifications does not alter the design or operation of any system or piece of equipment designed for the prevention or mitigation of accidents and transients. As a result, no new operating modes are introduced from which a new type accident becomes possible. Existing systems will continue to be operated per present design basis assumptions.

The proposed P/T limits were generated from the evaluation of the material capsule removed during the [s]pring Unit 1 outage of 1996. As a result, these limits include the latest available information on the reactor vessel materials. Furthermore, they will continue to be monitored per the requirements of the Technical Specifications and 10 CFR [Part] 50 Appendices G and H. For the above reasons, the changes do not create the possibility of a new type of accident.

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

The purpose of the P/T limits is to avoid a brittle fracture of the reactor vessel. As such, material capsules are removed periodically to determine the effects of neutron irradiation on reactor vessel materials. This change to the Unit 1 P/T curves is proposed to incorporate the evaluation results of the latest capsule removed during the [s]pring Unit 1 outage of 1996. Accordingly, these curves represent the latest information available on the reactor vessel materials. Also, the curves were generated using the approved methodologies of 10 CFR [Part] 50 Appendix G.

The pressure test curve (Figure 3.4.9-1) is also being revised to reflect exposure dependencies. These curves were generated for exposures of 16, 18, 20, 24, 28, and 32 EFPY [effective full-power year]. As previously described, each of these curves were generated using approved methodologies and all reflect the results of this latest material capsule report.

The proposed change does not affect the evaluation of any FSAR Unit 1 Chapter 14 transient and accident. Furthermore, the proposed change does not affect the operation of systems or equipment important to safety.

The Limiting Condition for Operation of Specification 3.4.9 will not change. Also, no Technical Specification surveillances or surveillance frequencies are revised as a result of this Technical Specification submittal, besides the fact that the P/T surveillances will now refer to the revised curves. Procedures regarding the monitoring of the P/T limits during reactor startup, cooldown, and leakage testing will not change as a result of this proposed Technical Specification change with respect to frequency of the surveillance or the methods used to perform the surveillances. Thus, the P/T limits will continue to be surveilled as before per the same procedures and the same frequencies.

No other Technical Specifications are affected by the proposed revision. The margin of safety to any Technical Specifications safety limit therefore is not reduced.

For the above reasons the new curves do not represent 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: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

NRC Project Director: Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: May 30, 1997

Description of amendment request: The proposed amendments would revise power sources to valves associated with low pressure coolant injection (LPCI) mode of residual heat removal (RHR) system.

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

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The LPCI valves operate to establish and maintain adequate core cooling following a LOCA [loss-of-coolant accident]. The proposed changes do not alter the function or mode of operation of the LPCI valves. Therefore, the probability of the LOCA accident is not increased. An analysis which considered the consequences of the various transients and accidents with the proposed change in power supply of the LPCI valves indicates the consequences are not increased.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously analyzed. The change in power supply to the LPCI valves maintains the original design criteria that a power supply independent of the remaining RHR subsystem be utilized for single-failure criteria. The function of the LPCI valves and any other existing equipment is not altered. Operation of the valves in the proposed configuration was analyzed, and no new failure modes exist. An analysis of the impact on the operation and design of other systems and components indicates no new failure modes are introduced. Therefore, these changes do not contribute to a new or different type of accident.

3. The proposed changes do not involve a significant reduction in the margin of safety. The change in power supply to the LPCI valves was evaluated relative to RHR and electrical distribution system function during normal and accident conditions. The proposed change does not alter the performance of any system safety functions. The results of the SAFER-GESTR LOCA analysis reconfirm the large margins existing in fuel peak cladding temperature under the proposed configuration. Therefore, there is no significant reduction in the margin of safety.

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

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

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

NRC Project Director: Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: June 13, 1997

Description of amendment request: The proposed amendments would revise the Technical Specification Limiting Condition for Operation 3.4.10 Pressurizer Safety Valves. Specifically, the change would reduce the nominal set pressure by 1 percent to 2460 pounds per square inch gauge (psig) and increase the tolerance to plus or minus 2 percent.

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

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

The increase in the PSV [pressurizer safety valve] tolerance from [plus or minus] 1% with a setpoint of 2485 psig to [plus or minus] 2% and reduction in the nominal setpoint from 2485 psig to 2460 psig has the net effect of reducing the minimum lift setting allowed by the TS [technical specifications] from 2460 psig to 2410 psig. The effects of this change have been evaluated for its impact on the assumed frequency of safety valve challenges and failures to reclose, and the proposed change was found to have a negligible impact. In other words, reducing the minimum lift setting does not significantly increase the probability of an inadvertent actuation of a safety valve during normal operation. Reducing the minimum lift setting does increase the potential that the PSVs may open during an event, but this change has been evaluated and does not adversely impact the consequences of any accident previously evaluated. No change to any equipment response or accident mitigation scenario has resulted, and there are no additional challenges to fission product barrier integrity. Therefore, the proposed change does not significantly increase the probability or consequences of any accident previously evaluated.

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

The increase in the PSV tolerance from [plus or minus] 1% with a setpoint of 2485 psig to [plus or minus] 2% and reduction in the nominal setpoint from 2485 psig to 2460 psig does not create the possibility of a new or different kind of accident than any accident previously evaluated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed change. The proposed revision to Technical Specification 3.4.10 does not challenge the performance or integrity of any safety-related systems. Therefore, the possibility of a new or different kind of accident is not created.

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

The proposed change to Technical Specification 3.4.10 does not involve a significant reduction in a margin of safety. The modification will have no affect on the availability, operability or performance of the safety-related systems and components. The increased PSV set pressure tolerance has been reviewed with respect to the accident analysis assumptions and requirements and evaluated or analyzed, as required. These evaluations and analyses determined that all applicable acceptance criteria continue to be met, thus the proposed increase in the PSV set pressure tolerance will not result in a significant reduction in the margin of safety associated with the acceptance criteria for the accident analyses.

The Bases of the Technical Specifications rely in part on the ability of the regulatory criteria being satisfied assuming the limiting conditions for operation for various systems. Conformance to the regulatory criteria for operation with the increased PSV set pressure tolerance is demonstrated, and the regulatory limits are not exceeded. Hence, the margin of safety as defined in the Bases for the Technical Specifications is not significantly reduced.

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

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

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia 30830

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308

NRC Project Director: Herbert N. Berkow

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

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

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

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. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

A. Revision to the Unit 2 Core Safety Limits

Analyses of reactor core safety limits are required as part of reload calculations for each cycle. TU Electric has performed the analyses of the Unit 2, Cycle 4 core configuration to determine the reactor core safety limits. The methodologies and safety analysis values result in new operating curves which, in general, permit plant operation over a similar range of acceptable conditions. This change means that if a transient were to occur with the plant operating at the limits of the new curve, a different temperature and power level might be attained than if the plant were operating within the bounds of the old curves. However, since the new curves were developed using NRC approved methodologies which are wholly consistent with and do not represent a change in the Technical Specification BASES for safety limits, all applicable postulated transients will continue to be properly mitigated. As a result, there will be no significant increase in the consequences, as determined by accident analyses, of any accident previously evaluated.

B. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints

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

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

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

SUMMARY

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

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

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

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

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

A. Revision to the Unit 2 Reactor Core Safety Limits

The TU Electric reload analysis methods have been used to determine new reactor core safety limits. All applicable safety analysis limits have been met. The methods used are wholly consistent with Technical Specification BASES 2.1 which is the bases for the safety limits. In particular, the curves assure that for Unit 2, Cycle 4, the calculated DNBR is no less than the safety analysis limit and the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The acceptance criteria remains valid and continues to be satisfied; therefore, no change in a margin of safety occurs.

B. Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints

Because the reactor core safety limits for CPSES Unit 2, Cycle 4 are recalculated, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint which protect the reactor core safety limits must also be recalculated. The Overtemperature N-16 reactor trip setpoint helps prevent the core and Reactor Coolant System from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. However, it was shown in these calculations that the current Unit 2 overtemperature reactor trip setpoint (presented in the current Technical Specifications and excluding the f(delta I) trip reset function) remains valid. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the Overtemperature reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). This event has been analyzed with the new safety analysis value for the Overtemperature reactor trip setpoint to demonstrate compliance with event specific acceptance criteria. Because all event acceptance criteria are satisfied, there is no degradation in a margin of safety.

The nominal Reactor Trip System instrumentation setpoints values for the Overtemperature N-16 reactor trip setpoint (Technical Specification Table 2.2-1) are determined based on a statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. The total uncertainty plus additional margin is applied in a conservative direction to the safety analysis trip setpoint value to arrive at the nominal and allowable values presented in Technical Specification Table 2.2-1. Meeting the requirements of Technical Specification Table 2.2-1 assures that the Overtemperature reactor trip setpoint assumed in the safety analyses remains valid. The CPSES Unit 2, Cycle 4 Overtemperature reactor trip setpoint is not significantly different from the previous cycle, and thus provides operational flexibility to withstand mild transients without initiating automatic protective actions. Although the value of the f(delta I) trip reset function setpoint is different, the Reactor Trip System instrumentation setpoint values for the Overtemperature N-16 reactor trip setpoint are consistent with the safety analysis assumptions which have been analytically demonstrated to be adequate to meet the applicable event acceptance criteria. Thus, there is no reduction in a margin of safety.

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

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

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

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

NRC Project Director: James W. Clifford, Acting

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

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

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

Commonwealth Edison Company, Docket No. STN 50-455, Byron Station, Unit No. 2, Ogle County, Illinois Docket No. STN 50-457, Braidwood Station, Unit No. 2, Will County, Illinois

Date of amendment request: May 24, 1997

Description of amendment request: The amendments revise the technical specifications related to venting of the emergency core cooling system pumps and associated piping. The application originally included Byron, Unit 1. However, on May 31, 1997, ComEd supplemented the application to request an emergency license amendment for Byron, Unit 1. Amendment No. 90 was issued on June 1, 1997.

Date of publication of individual notice in **Federal Register:** June 10, 1997 (62 FR 31633)

Expiration date of individual notice: July 10, 1997

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

Niagara Mohawk Power Corporation, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of application for amendment: May 16, 1997

Brief description of amendment: The proposed amendment would make an administrative change to add a supervisory position to the list of personnel who may be required to hold a senior reactor operator license. Date of publication of individual notice in **Federal Register:** June 4, 1997 (62 FR 30625)

Expiration date of individual notice: July 7, 1997

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

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 20, 1997, with the proposed no significant hazards consideration submitted by letter dated January 30, 1997, as supplemented February 27, April 11, May 14, and June 20 (2 letters), 1997

Brief description of amendment: The amendment authorizes Boston Edison Company (BECo) to change the UHS administrative limit from 68°F to 75 °F, and change the Updated Final Safety Analysis Report (UFSAR) to reflect the use of containment pressure to compensate for the deficiency in NPSH following a design basis accident and increase the accident analysis design UHS temperature from 65°F to 75°F. As part of this amendment, BECo has proposed to submit a Technical Specification amendment for the UHS temperature by the first quarter of 1998. In addition, within 180 days of issuance of this amendment, BECo has committed to complete the containment analysis using the ANS 5.1-1979 Decay Heat Curve with a 2-sigma uncertainty added. The staff considers BECo's commitments acceptable and has conditioned the amendment accordingly.

Date of issuance: July 3, 1997

Effective date: July 3, 1997

Amendment No.: 173

Facility Operating License No. DPR-35: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register:** February 26, 1997 (62 FR 8792) The February 27, April 11, May 14, and June 20 (2 letters), 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination as submitted by letter dated January 30, 1997. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 3, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 14, 1997, as supplemented May 16, and June 17, 1997

Brief description of amendment: The amendment approves changes to the Final Safety Analysis Report to reflect new analysis of the radiological consequences of dropping a fuel cask.

Date of issuance: June 26, 1997

Effective date: June 26, 1997

Amendment No. 73

Facility Operating License No. NPF-63. Amendment revises the Final Safety Analysis Report.

Date of initial notice in **Federal Register:** April 9, 1997 (62 FR 17226). The May 16, and June 17, 1997 supplemental information did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 1997. No significant hazards consideration comments received: No

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

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

Date of amendment request: April 11, 1997

Brief description of amendment: The amendment changes the Waterford steam Electric Station, Unit 3, Technical Specifications (TSs) by revising TS 3.6.2.2 and Surveillance Requirement 4.6.2.2 for the Containment Cooling System. Also, a Surveillance Requirement is added to verify that valves actuate on a Safety Injection Actuation Signal. To support this addition, Technical Specification Bases 3/4.3.6.2.2 is also included.

Date of issuance: July 3, 1997

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

Amendment No.: 131

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

Date of initial notice in **Federal Register:** April 22, 1997 (62 FR 19626) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 3, 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

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

Date of application for amendment: April 17, 1997

Brief description of amendment: The amendment modifies Technical Specification 3.7.14 by clarifying the actions to be taken when an area temperature exceeds its temperature limit.

Date of issuance: June 24, 1997

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

Amendment No.: 141

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

Date of initial notice in **Federal Register:** (62 FR 27798 May 21, 1997) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 24, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 15, 1997

Brief description of amendment: The amendment makes changes to Technical Specification (TS) Sections 4.3.3.6 and 4.6.4.1, which require that the hydrogen monitors be periodically tested. Specifically, the changes increase the testing interval of the monitor's hydrogen sensor, correct inconsistencies between the TS surveillances, and make changes to the Bases of the surveillances.

Date of issuance: June 24, 1997

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

Amendment No.: 142

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

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27797) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 24, 1997. No significant hazards consideration comments received: No.

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

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: April 11, 1997

Brief description of amendments: These amendments revise Technical Specification (TS) 3/4.6.2.3, ''Containment Cooling System,'' and its associated Bases section to ensure that the TSs properly test the containment fan cooling units' post-accident mode of operation.

Date of issuance: June 24, 1997

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

Amendment Nos. 197 and 180

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

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27799) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 24, 1997. No significant hazards consideration comments received: No.

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

Tennessee Valley Authority, Docket Nos. 50-327 and 50328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: March 13, 1997, as supplemented on June 26, 1997 (TS 97-01)

Brief description of amendments: The amendments change the Technical Specifications by raising the allowable U-235 enrichment, as specified in Section 5.6.1.2, of fuel stored in the new fuel pit storage racks from 4.5 to 5.0 weight percent.

Date of issuance: July 1, 1997

Effective date: July 1, 1997

Amendment Nos.: 225 and 216

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

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27802). The June 26, 1997 supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in an environmental assessment dated June 16, 1997, and a Safety Evaluation dated July 1, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 27, 1993, as supplemented by letters dated November 9, 1993, April 26, 1996, and September 25, 1996

Brief description of amendment: The amendment revises the Technical Specifications to incorporate the revised 10 CFR Part 20, Standards for Protection Against Radiation.

Date of issuance: June 19, 1997

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

Amendment No.: 151

Facility Operating License No. DPR-28. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 507) The November 9, 1993, April 26, 1996, and September 25, 1996, submittals 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 June 19, 1997. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: June 4, 1996 (TSCR 188 and 189), as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997

Brief description of amendments: These amendments revise Technical Specifications (TS) 15.1, ''Definitions;'' TS 15.2.1, ''Safety Limit, Reactor Core;'' TS 15.2.3, ''Limiting Safety System Settings, Protective Instrumentation;'' TS 15.3.1, ''Reactor Coolant System,'' Section C, ''Maximum Coolant Activity,'' and Section G, ''Operational Limitations;'' TS 15.3.4, ''Steam and Power Conversion System;'' TS 15.3.5, ''Instrumentation System;'' TS 15.4.1, ''Operational Safety Review;'' TS 15.5.3, ''Design Features-Reactor;'' and TS 15.6.9, ''Plant Reporting Requirements'' to reflect parameters associated with new steam generators in Unit 2 and changes in analyses that affect both Units 1 and 2.

Date of issuance: July 1, 1997

Effective date: July 1, 1997. The TS shall be implemented within 45 days from the date of issuance and the Final Safety Analysis Report changes shall be implemented by June 30, 1998. Implementation of these amendments includes incorporation of accident analyses submitted in support of this amendment into the Final Safety Analysis Report in sufficient detail to support future evaluations performed in accordance with 10 CFR 50.59 and as described in the licensee's applications dated June 4, 1996, as supplemented on August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 20, and April 2, 1997, and evaluated in the staff's safety evaluation dated July 1, 1997.

Amendment Nos.: 173, 177

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

Date of initial notice in **Federal Register:** July 3, 1996 (61 FR 34903 and 61 FR 34904) and April 9, 1997 (62 FR 17243) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 1, 1997. No significant hazards consideration comments received: No.

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

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

Date of amendment request: March 21, 1997, as supplemented by letter dated April 15, 1997

Brief description of amendment: The amendment revises Technical Specification 6.8.5.b to provide an exception to the examination requirements of Regulatory Guide 1.14, Revision 1, ''Reactor Coolant Pump Flywheel Integrity'' and delays the inspection of the ''D'' reactor coolant pump flywheel to the Fall 1997 refueling outage. A typographical error in TS 6.8.5.c is corrected.

Date of issuance: June 24, 1997

Effective date: June 24, 1997, to be implemented within 30 days of issuance.

Amendment No.: 106

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

Date of initial notice in **Federal Register:** May 21, 1997 (62 FR 27803) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 24, 1997. No significant hazards consideration comments received: No.

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

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration and opportunity for a hearing (Exigent Public Announcement Or Emergency Circumstances)

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

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

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

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

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

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

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By August 15, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, and to the attorney for the licensee.

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

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

Date of application for amendment: December 11, 1996, as supplemented March 27, 1997, April 17, 1997, and June 17, 1997

Brief description of amendment: The amendment revises Technical Specifications to allow extended rod position indicator deviation limits, on-line calibration of the rod position indication and to clarify the operability requirements during calibration.

Date of issuance: June 27, 1997

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

Amendment No.: 194

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on June 25, 1997. The notice was published in the Peekskill Evening Star on June 20-25, 1997.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of New York and final no significant hazards consideration determination are contained in a Safety Evaluation dated June 27, 1997.

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

North Atlantic Energy Service Corporation, Dockets Nos. 50-443, Seabrook Station, Unit 1, Seabrook, Massachusetts

Date of amendment request: June 19, 1997

Brief description of amendment: The amendment revised Technical Specification 6.8.1.6.b. to include a reference to the NRC-approved Westinghouse Topical Report WCAP-12610-P-A, ''VANTAGE+ Fuel Assembly Reference Core Report,'' dated April 1995.

Date of issuance: June 24, 1997

Effective date: As of the date of issuance, and to be implemented before transition into Operational Mode 2 during startup from Refueling Outage 5.

Amendment No.: 52

Facility Operating License No. NPF-86: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, consultation with the States of New Hampshire and Massachusetts, and final no significant hazards considerations determination are contained in the safety evaluation dated June 24, 1997.

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

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141-0270 Acting

NRC Project Director: Patrick D. Milano

North Atlantic Energy Service Corporation, Dockets Nos. 50-443, Seabrook Station, Unit 1, Seabrook, Massachusetts

Date of amendment request: May 29, 1997

Brief description of amendment: The amendment modifies Technical Specification 5.3.1 by replacing the current term ''zircaloy'' with terminology that explicitly identifies the NRC-approved Westinghouse fuel assembly design in use at the Seabrook Station consisting of assemblies with either ZIRLO or Zircaloy-4 fuel cladding material.

Date of issuance: June 24, 1997

Effective date: As of the date of issuance, and to be implemented before transition into Operational Mode 2 during startup from Refueling Outage 5.

Amendment No.: 53

Facility Operating License No. NPF-86: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on June 10, 1997. The notice was published in Foster's Daily Democrat and in the Portsmouth Herald on June 4, 1997. Public comments were received, and they have been addressed in the staff's safety evaluation.

The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the States of New Hampshire and Massachusetts, and final no significant hazards determination are contained in a safety evaluation dated June 24, 1997.

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

Attorney for licensee: Lillian M. Cuoco, Esquire, Northeast Utilities Service Company, Post Office Box 270, Hartford CT 06141-0270 Acting

NRC Project Director: Patrick D. Milano

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

Date of application for amendment: March 22, 1997, as supplemented by letters dated April 2, April 3, April 9, April 15, and May 14, 1997. Additional information was also received by telefax on May 19, 1997.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.3.1.1.15, Reactor Protection System (RPS) Response Time functions 3 and 4 and SR 3.3.6.1.7, Primary Containment Isolation System Response Time, functions 1.a, 1.b, and 1.c, adding a note to indicate that the sensor is excluded from response time testing when verifying that the response time is within limits. The amendment also revises SR 3.3.5.1.7, Emergency Core Cooling System (ECCS) Response Time by relocating the requirements to SR 3.5.1.8, ECCS Operating, and adding a note to SR 3.5.1.8 to indicate that no actuation instrumentation response time measurement is required. Additionally, SR 3.5.1.8 requires that the SR be met in MODES 1, 2, and 3, whereas the previous SR 3.3.5.1.7 was required to be met in MODES 1, 2, 3, 4, and 5.

Date of Issuance: June 11, 1997

Effective date: June 11, 1997

Amendment No.: 150

Facility Operating License No. NPF-21. The amendment revised the Technical Specifications. Press release issued requesting comments as to proposed no significant hazards consideration: Yes. April 11, 1997. Tri-City Herald (Washington). Comments received: No. The Commission's related evaluation of the amendments, finding of exigent circumstances, consultation with the State of Washington and final determination of no significant hazards consideration are contained in a Safety Evaluation dated June 11, 1997.

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

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

NRC Project Director: William H. Bateman

Dated at Rockville, Maryland, this 9th day of July 1997.

For the Nuclear Regulatory Commission

Elinor G. Adensam,

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

[Doc. 97-18513 Filed 7-15-97; 8:45 am]

BILLING CODE 7590-01-F

PENSION BENEFIT GUARANTY CORPORATION

Agency Information Collection Activities; OMB Approval Received; Disclosure of Premium-Related Information

AGENCY: Pension Benefit Guaranty Corporation.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act, this notice announces the Office of Management and Budget's approval of a collection of information contained in the Pension Benefit Guaranty Corporation's final rule amending its premium payment regulation.

FOR FURTHER INFORMATION CONTACT: Harold J. Ashner, Assistant General Counsel, or James L. Beller, Attorney, Pension Benefit Guaranty Corporation, Office of the General Counsel, Suite 340, 1200 K Street, NW., Washington, DC 20005–4026, 202–326–4024 (202–326–4179 for TTY and TDD).

SUPPLEMENTARY INFORMATION: On July 9, 1997, the PBGC published in the **Federal Register** (62 FR 36663) a final rule amending its premium payment regulation to provide for submission to the PBGC of plan records that are necessary to support premium filings. This rule contains information collection requirements. On July 11, 1997, OMB approved the collection of information requirements with respect to this final rule under OMB control number 1212–0009 (expires February 28, 1998). An agency may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

Issued in Washington, D.C. this 11th day of July, 1997.

John Seal,

Acting Executive Director, Pension Benefit Guaranty Corporation.

[FR Doc. 97–18720 Filed 7–15–97; 8:45 am]

BILLING CODE 7708–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension: Form 2–E and Rule 609, SEC File No. 270–222, OMB Control No. 3235–0233; Rule 6c–7, SEC File No. 270–269, OMB Control No. 3235–0276; and Rule 11a–2, SEC File No. 270–267, OMB Control No. 3235–0272.

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities And Exchange Commission (''Commission'') is soliciting comments on the collections of information summarized below. The Commission plans to submit these existing collections of information to the Office of Management and Budget for extension and approval.

Form 2–E is used, pursuant to Rule 609 of Regulation E under the Securities Act of 1933, by small business investment companies or business development companies engaged in limited offerings of securities to report semi-annually the progress of an offering, including the number of shares sold. The form solicits information such as the dates an offering has commenced and completed, the number of shares sold and still being offered, amounts received in the offering, and expenses and underwriting discounts incurred in the offering. This information assists the Commission staff in determining whether the issuer has stayed within the limits of an exemptive offering.