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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 located at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Dated at Rockville, MD, this 31st day of May 1996.

For the Nuclear Regulatory Commission.

George F. Wunder,

Project Manager, Project Directorate I–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

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Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 11, 1996, through May 23, 1996. The last biweekly notice was published on May 22, 1996 (61 FR 25696).

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

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

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

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

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

By July 5, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment will relocate the administrative controls related to the quality assurance review and audit requirements of Section 6 from the Pilgrim Station Technical Specifications to the Boston Edison Quality Assurance Manual. This change is in accordance with the guidance contained in NRC Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance."

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

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

The change will relocate the administrative controls related to the quality assurance review and audit requirements from the technical specifications to the quality assurance plan. These changes are administrative in nature and do not impact initiators of analyzed events, accident mitigation capabilities, or transient events. The quality assurance program is a logical candidate for such relocation due to the controls imposed by such regulations as Appendix B to 10 CFR [Part] 50, the existence of NRC approved quality assurance plans and commitments to industry quality assurance standards, and the established quality assurance program change control process in 10 CFR 50.54(a). Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The change will relocate the administrative controls related to the quality assurance review and audit requirements from the technical specifications to the quality assurance plan. The quality assurance program is a logical candidate for such relocation due to the controls imposed by such regulations as Appendix B to 10 CFR [Part] 50, the existence of NRC approved quality assurance plans and commitments to industry quality assurance standards, and the established quality assurance program change control process in 10 CFR 50.54(a). The proposed changes do not involve a physical alteration of the plant or changes in methods governing plant operation. The changes will not impose or eliminate any new or different requirements. Therefore the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The change will relocate the administrative controls related to the quality assurance review and audit requirements from the technical specifications to the quality assurance plan. These changes are administrative in nature. The quality assurance program is a logical candidate for such relocation due to the controls imposed by such regulations as Appendix B to 10 CFR [Part] 50, the existence of NRC approved quality assurance plans and commitments to industry quality assurance standards, and the established quality assurance program change control process in 10 CFR 50.54(a). The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Therefore, the operation of PNPS [Pilgrim Nuclear Power Station] in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Jocelyn A. Mitchell, Acting

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

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment will reflect the implementation of 10 CFR Part 50, Appendix J, Option B at the Pilgrim Nuclear Power Station.

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 operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes do not involve any physical or operational changes to structures, systems or components. The proposed changes provide a mechanism within the TS [Technical Specifications] for implementing a performance-based leakage rate test program which was promulgated by the revision to 10CFR50 to incorporate Option B into Appendix J. The TS Limiting Conditions for Operation (LCO) remain unaffected by these changes. Thus, the safety design basis for the accident mitigation functions of the primary containment is maintained. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

2. The operation of Pilgrim Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Revising surveillance requirement acceptance criteria and frequencies does not physically modify the plant and does not modify the operation of any existing equipment. Further, the TS LCOs remain unaffected by these changes.

3. The operation of Pilgrim Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant reduction in the margin of safety, nor do they affect a safety limit, an LCO, or the manner in which plant equipment is operated. The NRC letter dated November 2, 1995, recognizes that changes similar to the proposed changes are required to implement Option B of 10CFR50, Appendix J. In NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' which forms the basis for the Appendix J revision, the NRC concludes that adoption of performance-based test intervals for Appendix J testing will not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Plymouth Public Library, 11 North Street, Plymouth, Massachusetts 02360.

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

NRC Project Director: Jocelyn A. Mitchell, Acting

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

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment would modify the definition of ''Core Alteration,'' and the Limiting Condition for Operation, Surveillance conditions and Bases section associated with Technical Specification (TS) 3.7.C, ''Secondary Containment.''

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

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

Operation of PNPS [Pilgrim Nuclear Power Station] in accordance with the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

Proposed Change ι1: Definition of ''Alteration of the Reactor Core≥

The definition, ''Alteration of the Reactor Core'', is being revised so that the term will apply only to those activities that create the potential for a reactivity excursion and, therefore, warrant special precautions or controls in the TS. The proposed definition includes normal control rod movement in the definition, but excludes control rod drive movement (such as rod removal from the core) when all four fuel bundles surrounding a control rod are removed. The proposed change does not increase the probability or consequences of an accident because the proposed definition, by identifying activities with the potential for causing a reactivity excursion, ensures that the additional precautions and controls in the TS are implemented at all appropriate times. In addition, the movement of components excluded by this definition is not assumed in the initiation of any analyzed event. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change ι2: Secondary Containment

The current specifications are revised to specify more clearly when secondary containment is required, what actions to take if secondary containment is inoperable, and time frames for completing the actions. These revisions enhance the existing specification and serve to make it more definitive by encompassing the conditions currently specified by TS and supplementing them to specify other conditions when secondary containment is required.

Surveillances 4.7.C.1.a and b were only necessary during initial and Cycle 1 operations. Removing obsolete information from the existing specifications, re-numbering and re-arranging the wording is an administrative change.

These changes are administrative in nature and do not impact initiators of analyzed events, accident mitigation capabilities, or transient events. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The operation of PNPS in accordance with the proposed license amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated because of the following:

Proposed Change ι1: Definition of ''Alteration of the Reactor Core≥

The definition change specifies more accurately which component movements constitute a ''Core Alteration''. This change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will allow movement of some components (camera, lights, etc.) during times when ''Core Alterations'' have been halted since these components will not affect core reactivity. Removal of a control rod involves unlatching and withdrawal/insertion from over-vessel handling equipment. These activities necessitate, by design, the removal of the adjacent four fuel assemblies. With this configuration (no fuel in the cell; handling the associated control rod), the proposed change will allow movement of a ''reactivity control component'' while not imposing requirements unique to ''Core Alterations'' (note: other requirements, such as those for handling loads over irradiated fuel, will remain applicable). The reactivity effects of this control rod movement are more than compensated for by the initial removal of the fuel assemblies. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change ι2: Secondary Containment

The proposed change does not eliminate or relax any existing TS condition. Rather, it better defines when secondary containment is required, provides action statements for inoperability and removes obsolete requirements (from first operating cycle). This change does not involve a physical change to structures, systems or components, and the safety design bases for the accident mitigating function of the secondary containment is maintained. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The operation of PNPS in accordance with the proposed license amendment will not involve a significant reduction in a margin of safety because of the following:

Proposed Change ι1: Definition of ''Alteration of the Reactor Core≥

The proposed definition more accurately identifies those activities with the potential for causing a reactivity excursion. The more accurate identification of ''Core Alterations'' will ensure that when there is a potential for reactivity excursions, appropriate precautions are applied. The components now excluded from the proposed definition are those that do not have the capability for adversely impacting core reactivity. The proposed change has no impact on safety analysis assumptions. Therefore, the change will not involve a significant reduction in a margin of safety.

Proposed Change ι2: Secondary Containment

The proposed additions of applicability conditions provide a more precise understanding of when secondary containment integrity is required and what actions to take if it becomes inoperable. The change does not eliminate any existing conditions. The deletion of surveillances applicable only for the first operating cycle and re-numbering and re-arranging the remaining surveillance wording is an administrative change and has no impact on the operation of the plant or mitigation of accidents. Therefore, the operation of the facility in accordance with this proposed amendment would not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Jocelyn A. Mitchell, Acting

Carolina Power & Light Company, et al., Docket No. 50-325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: April 8, 1996

Description of amendment request: The licensee has proposed to revise the Technical Specifications (TS) to include the following changes: 1. The Minimum Critical Power Ratio (MCPR) Safety Limit specified in TS 2.1.2 from 1.07 to 1.09 for Unit 1 Cycle 11 operation; TS 5.3.1 to reflect the new fuel type (GE13) that will be inserted during Unit 1 Refueling Outage 10; 2. The acceptable range of sodium pentaborate concentration for the standby liquid control system shown in TS Figure 3.1.5-1 to reflect changes to poison material concentration needed to achieve reactor shutdown based on the new GE13 fuel type.

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

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

Proposed Change 1

The proposed amendment will allow the loading and use of GE13 fuel assemblies in the Brunswick Unit 1 reactor core. The use of GE13 fuel assemblies requires that the safety limit minimum critical power ratio value also be revised. The safety limit minimum critical power ratio is established to maintain fuel cladding integrity during operational transients. The GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energy's reload licensing methodology Topical Report (NEDE-24011-P-A-11, ''General Electric Standard Application for Reactor Fuel (GESTAR II)'' dated November 1995).

The proposed revision of the safety limit minimum critical power ratio does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident. The change does not affect the design, materials, or construction standards applicable to the fuel bundles in a manner that could change the probability of an accident.

A methodology that has been previously reviewed and accepted by the Nuclear Regulatory Commission was used to derive both the existing and updated safety limit minimum critical power ratio value. The same methodology and criteria have been applied to derive the existing safety limit minimum critical power ratio of 1.07 as that used to derive the updated safety limit minimum critical power ratio value of 1.09. The updated safety limit minimum critical power ratio assures that fuel cladding protection equivalent to that provided with the existing safety limit minimum critical power ratio value is maintained. This ensures that the consequences of previously evaluated accidents are not significantly increased.

Proposed Change 2

The standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system. The standby liquid control system must be capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. Technical Specification Figure 3.1.5-1 specifies the acceptable range of concentrations and volumes for sodium pentaborate solution used as a neutron absorber (i.e., for reactivity control). The portion of the sodium pentaborate concentration range shown in Technical Specification Figure 3.1.5-1 applicable to the lower range of tank volumes is being revised to increase the required concentration of sodium pentaborate solution. This change is needed to account for the additional shutdown reactivity needed based on the planned use of GE13 fuel assemblies as reload fuel for the Unit 1 reactor core. Since the standby liquid control system is independent from the normal means of controlling reactor core reactivity and not used to control core reactivity during normal plant operations, the proposed revision to the sodium pentaborate concentration curve for the standby liquid control system does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident.

The current volume-concentration range of sodium pentaborate used in the standby liquid control system will achieve a sufficient concentration of boron in the reactor vessel to ensure reactor shutdown. Based on the increased reactivity of the new GE13 reload fuel assemblies, the required sodium pentaborate volume-concentration range is being revised to ensure sufficient neutron absorbing solution is available to achieve reactor shutdown; therefore, the consequences of an accident previously evaluated are not significantly increased.

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

Proposed Change 1

The GE13 fuel assembly has been designed and complies with the acceptance criteria contained in General Electric Nuclear Energy's standard application for reactor fuel (GESTAR-II), which provides the latest acceptance criteria for new General Electric fuel designs. The GE13 fuel assembly complies with GESTAR-II acceptance criteria that have been previously reviewed and accepted by the Nuclear Regulatory Commission. The similarity of the GE13 fuel design to the previously accepted GE11 fuel design, in conjunction with the increased critical power capability of the GE13 fuel design, ensure that no new mode or condition of plant operation is being authorized by the loading and use of the

GE13 fuel type. The proposed revision of the safety limit minimum critical power ratio from 1.07 to 1.09 does not modify any plant controls or equipment that will change the plant's responses to any accident or transient as given in any current analysis. Therefore, the proposed change to allow the loading and use of the GE13 fuel type and the revision of the safety limit minimum critical power ratio value from 1.07 to 1.09 will not create the possibility for a new or different kind of accident from any accident previously evaluated.

Proposed Change 2

As discussed above, the standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system and is capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. The proposed revision to the sodium pentaborate concentration range does not modify the standby liquid control system or its controls, does not modify other plant systems and equipment, and does not permit a new or different mode of plant operation. As such, the proposed revision to the minimum pentaborate concentration value does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Proposed Change 1

As previously discussed, the GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energy's reload licensing methodology Topical Report (NEDE-24011-P-A-11, "General Electric Standard Application for Reactor Fuel (GESTAR II)" dated November 1995). The safety limit minimum critical power ratio value is selected to maintain the fuel cladding integrity safety limit (i.e., that 99.9 percent of all fuel rods in the core are expected to avoid boiling transition during operational transients). Appropriate operating limit minimum critical power ratio values are established, based on the safety limit minimum critical power ratio value, to ensure that the fuel cladding integrity safety limit is maintained. The operating limit minimum critical power ratio values are incorporated in the Core Operating limits Report as required by Technical Specification 6.9.3.1. The new GE13 safety limit minimum critical power ratio value of 1.09 is based on the same fuel cladding integrity safety limit criteria [as] that for the GE11 safety limit minimum critical power ratio value of 1.07 (i.e., that 99.9 percent of all fuel rods in the core are expected to avoid boiling transition during operational transients); therefore, the proposed change does not result in a significant reduction in the margin of safety.

Proposed Change 2

As previously stated, the purpose of the standby liquid control is to inject a neutron absorbing solution into the reactor in the event that a sufficient number of control rods cannot be inserted to maintain subcriticality. Sufficient solution is to be injected such that the reactor will be brought from maximum rated power conditions to subcritical over the entire reactor temperature range from maximum operating to cold shutdown conditions. General Electric methodology establishes a fuel type dependent standby liquid control system shutdown margin to account for calculational uncertainties. General Electric calculations show that an in-vessel concentration of 660 ppm will provide a standby liquid control system minimum shutdown margin in excess of the 3.2%[delta]k value required for the GE13 fuel. To achieve an in-vessel concentration of 660 ppm, the acceptable range of standby liquid control system tank concentrations is being revised for the lower range of tank volumes. Thus, the proposed revision of the standby liquid control system sodium pentaborate volume-concentration range ensures that there will not be a significant reduction in the amount of available shutdown margin and, therefore, not a significant reduction in the margin of safety.

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

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

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: February 27, 1996

Description of amendment request: The proposed license amendment would modify the Action Statement of Technical Specification (TS) 3.7.1.1.1. Currently, the TS action statement requires that with the self actuation function on one or more main steam line code safety valves associated with an operating loop inoperable, the licensee must restore the inoperable valve to operable status within 4 hours. Otherwise, the plant must be in hot standby within the next 6 hours and in hot shutdown within the following 30 hours. The proposed change will allow continued power operation at reduced power levels with main steam safety valves inoperable. The proposed change is consistent with the philosophy of the Westinghouse Standard Technical Specifications.

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

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

The proposed change to the Action Statement of LCO [Limiting Condition for Operation] 3.7.1.1.1 will allow indefinite operation at less than or equal to 75% power in the event that the self actuation function of no more than one safety valve per steam generator is inoperable, and allow indefinite operation at less than or equal to 50% power in the event that the self actuation function of no more than two safety valves per steam generator is inoperable. The requirement to reduce power will ensure that there is no increase in the consequences of a loss of load accident. The proposed change is consistent with the methodology in the Westinghouse Standard Technical Specifications. The methodology is conservative, since the PORVs [power operated relief valves] cannot affect the time of reactor trip on high pressurizer pressure. Thus, it is concluded that the change does not increase the consequences of any previously evaluated accident.

The change only specifies a power reduction in the event that the self actuation function of steam generator safety valves is inoperable. It does not affect the probability of any accident. The change by itself does not affect the likelihood of an inoperable safety valve.

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

The change only specifies a power reduction in the event that the self actuation function of steam generator safety valves is inoperable. This does not create the potential for a new or different kind of accident. The lower power level assures that peak steam generator pressure and RCS [reactor coolant system] pressure will remain below 110% of design. This provides assurance that no equipment failure will occur due to overpressurization. Thus, the change does not create the possibility for a new or different kind of accident.

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

The allowable power levels have been selected, consistent with the Westinghouse Standard Technical Specifications, to assure that steam generator and RCS pressure will remain below 110% of design. Thus, there is no reduction in a margin of safety for overpressure protection.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: March 7, 1996

Description of amendment request: The licensee will be replacing a locally operated (manual) containment sump suction isolation valve, RH-V-808A, with a remote manually operated (motor operated) valve, RH-MOV-808A during the upcoming Cycle 19 refueling outage. As a result, changes are being requested to the Haddam Neck Plant Technical Specifications to reflect this design change.

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

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

The proposed technical specification change to Section 3/4.4.6.2 and its bases are the replacement of the designation RH-V-808A with RH-MOV-808A. There are no changes to the requirements of this specification and this change is therefore an administrative change. The changes to Section 3/4.5.1 will make the requirements for RH-MOV-808A identical to those of RH-MOV-22. RH-V-808A is being converted to a motor operated valve (MOV). This MOV will make the ability to establish a suction path from the containment to the Residual Heat Removal (RHR) System single failure proof from the control room. Both RH-MOV-22 and RH-MOV-808A will be opened to establish containment sump recirculation post-loss of coolant accident (LOCA). This will provide added assurance that core cooling will be maintained in the switch from injection to containment sump recirculation following a LOCA. The requirement for RH-MOV-808A to be closed and its hand wheel locked can not cause an accident. The credit for operation of RH-MOV-808A to ensure that the establishment of containment sump recirculation is single failure proof is equivalent to the current crediting of RH-V-808A with the only difference being that operation of the valve can now be performed from the control room. Also, since both RH-MOV-22 and RH-MOV-808A will be procedurally opened during establishment of containment sump recirculation, the elimination of the requirement to lock open the breaker for RH-MOV-22 will not affect the consequences of a LOCA. The proposed changes that reflect the conversion of RH-V-808A to a MOV and the proposed changes in how the valve is used do not increase the consequences of a LOCA.

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

The proposed changes will require RH-MOV-808A to be closed with the hand wheel locked. This provides assurance that the valve is in the required position. Also, RH-MOV-808A will be capable of remote manual operation during the monthly surveillance which provides assurance that the valve can be repositioned if necessary. The proposed opening of RH-MOV-808A at the same time as RH-MOV-22 is opened, provides greater assurance that a suction path is available to the RHR pumps as well as lowering the total effective piping resistance from the containment sump to the pump suction. Therefore, the proposed changes do not introduce the possibility of a new or different kind of accident.

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

The proposed changes make RH-MOV-808A identical to RH-MOV-22 with the exception that RH-MOV-808A will not get a closure signal on Safety Injection Actuation. Both RH-MOV-22 and RH-MOV-808A are containment isolation valves in a closed system. For closed systems, the containment isolation requirement is that the valves be either: a) automatic, b) locked closed, or c) capable of remote manual operation. RH-MOV-808A and RH-MOV-22 are both capable of remote manual operation and therefore do not need automatic closure when they are opened as part of the technical specification required surveillance. Therefore, the proposed changes can not cause 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: Russell Library, 123 Broad Street, Middletown, CT 06457.

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: March 28, 1996

Description of amendment request: The proposed license amendment will add an additional footnote to Limiting Condition for Operation (LCO) 3.4.2.1 and revise an existing footnote for LCO 3.4.2.2. Currently, the footnote for LCO 3.4.2.2 requires the pressurizer code safety valve as-found lift setting to be within +3 percent and -1 percent of the setpoint. The proposed change will relax the negative as-found lift tolerance to -3 percent. The as-left lift tolerance will remain as plus or minus 1 percent. The same footnote will be added to LCO 3.4.2.1.

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

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

The proposed change will relax the pressurizer safety valve negative as-found lift tolerance to -3 percent. The as-left lift tolerance will remain as plus or minus 1 percent. This proposed technical specification change will allow for the full use of the plus or minus 3 percent as-found acceptance criterion for valve testing consistent with 1989 ASME Section XI, Subsection IWV. The relaxing of the as-found lift tolerance can not cause an accident. The relaxing of the tolerance will allow the safety valve setpoint to be closer to the Power Operated Relief Valve (PORV) setpoint and could result in a slightly lower pressure for overheating events. The analysis that takes credit for the increase in pressure to the PORV setpoint is the Loss of Load analysis. The minimum departure from nucleate boiling ratio (DNBR) was reanalyzed without taking any credit for the transient increase in pressure. The minimum DNBR still remains above the acceptance criterion as well as above the limiting minimum DNBR predicted for all Updated Final Safety Analysis Report Chapter 15 accidents. Also, the relaxed tolerance in conjunction with a lower safety valve blowdown, yet still conservative, results in a slightly higher average pressure for a valve lift/reset cycle. This means that pressurizer overfill will not be predicted for the limiting transient, loss of feedwater. Thus, the proposed relaxation of as-found lift tolerance does not increase the probability or consequences of the design basis accidents previously evaluated.

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

The proposed relaxation of the lift tolerance still requires the safety valve lift setpoint to be above both the PORV setpoint and the pressurizer high pressure reactor trip setpoint. In addition, the as-left setpoint is not being changed. The relaxed tolerance in combination with a conservative safety valve blowdown still will preclude the prediction of water relief from the pressurizer. This means that the proposed change does not introduce the possibility of a new or different kind of accident.

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

The proposed relaxation of the as-found lift tolerance for valve testing is consistent with 1989 ASME Section XI, Subsection IWV. The as-left lift tolerance will remain plus or minus 1 percent. In addition, the design basis analyses still meet their acceptance criteria with the -3 percent lift tolerance. Therefore, the proposed change can not cause 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

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

Date of amendment request: April 16, 1996

Description of amendment request: The licensee is proposing to revise the Technical Specifications to permit the Haddam Neck Plant to remain in Mode 1, 2, 3, or 4 with the average water temperature of the ultimate heat sink (UHS) greater than 90° additional action has been added which would require the plant to be placed in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours upon identifying that the average water temperature of the UHS is greater than 95°F. In addition, the licensee is proposing to include a new surveillance requirement for monitoring the average circulating water inlet temperature to be within its limits when the average water temperature of the UHS exceeds 89°F.

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

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

The proposed addition to the Action Statement of LCO 3.7.12 of an 8 hour period to monitor the average water temperature of the UHS does not involve an increase in the probability of an accident previously evaluated. The probability of an accident previously evaluated is not increased by a short-term increase in the average water temperature of the UHS. An evaluation of the service water loads associated with the loss-of-offsite power and a coincident worst case single failure of a diesel generator to start (resulting in the loss of two of the four service water pumps) determined that there is adequate margin to accomplish plant cooldown at a service water inlet temperature of 95°F. The recirculation phase of a LOCA [loss-of-coolant accident] was evaluated to verify that adequate flow would be available to the RHR [residual heat removal] heat exchangers. The most limiting assumptions for the recirculation phase are offsite power is available and one RHR heat exchanger service water isolation valve fails to open. The injection phase of a LOCA was evaluated to verify that adequate flow would be available to the CAR [containment air recirculation] fan cooling coils. The most limiting assumption for the injection phase is a loss-of-offsite power. The results of these evaluations determined that there is adequate service water flow to accomplish plant cooldown with average water temperature of the UHS up to 95°F. CYAPCO [Connecticut Yankee Atomic Power Company] also proposes to include an additional surveillance requirement to monitor the average water temperature of the UHS at least once per hour if the average water temperature of the UHS exceeds 89°F. This additional surveillance requirement ensures increased operator awareness as the average water temperature of the UHS approaches the 90°F LCO limit. Based on the above, there is no significant increase in the consequences of any accident previously evaluated.

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

The proposed technical specification changes do not create the possibility of a new or different kind of accident from those previously evaluated. The addition of an 8 hour time period to monitor the average water temperature of the UHS increases from 6 to 14 hours the amount of time that is allowed before the plant must proceed to Hot Standby should the average water temperature of the UHS increase above 90°F. This extension of the time allowed for the plant to be in Hot Standby does not change the plant configuration. CYAPCO also proposes to include an additional surveillance requirement to monitor the average water temperature of the UHS at least once per hour if the average water temperature of the UHS exceeds 89°F. This additional surveillance requirement ensures increased operator awareness as the average water temperature of the UHS approaches the 90°F LCO limit.

As such, the changes do not create the possibility of a new or different kind of accident from those previously evaluated.

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

The proposed technical specification changes do not involve a significant reduction in any margin of safety. The addition of an 8 hour time period to monitor the average water temperature of the UHS increases from 6 to 14 hours the time required before the plant must proceed to Hot Standby should the average water temperature of the UHS temperature [exceed] 90°F. An evaluation has been performed to demonstrate that the risk significance associated with the increased action time is very low. In addition, safe shutdown capability has been demonstrated for service water inlet temperatures as high as 95°F. The addition of a surveillance requirement to monitor the average water temperature of the UHS at least once per hour if the average water temperature of the UHS exceeds 89°F is an additional requirement, limitation, or restriction not currently within the technical specifications. Therefore, these 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: Russell Library, 123 Broad Street, Middletown, CT 06457

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

NRC Project Director: Phillip F. McKee

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

Date of amendment request: April 22, 1996

Description of amendment request: The proposed amendment will allow the use of the performance-based containment leakage testing requirements described in 10 CFR Part 50, Appendix J, Option B.

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

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

The changes involved in this license amendment request revise the testing criteria for the containment penetrations. The revised criteria will be based on the guidance in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' This guidance allows for the use of relaxed testing frequencies for containment penetrations that have performed satisfactorily on a historical basis. The Containment Leakage Rate Testing Program considers the type of service, the design of the penetration, and the safety impact of the penetration in determining the testing interval of each penetration. The NRC Staff has reviewed the potential impact of performance-based testing frequencies for containment penetrations during the development of the Option B regulation. The NRC Staff review is documented in NUREG-1493, ''Performance-Based Containment Leakage-Test Program.'' The review concluded that reducing the frequency of Type A tests (Integrated Leakage Rate Tests) from three per 10 years to one per 10 years leads to an imperceptible increase in risk. For Type B and C testing (Local Leakage Rate Tests), the change in testing frequency should not have significant impact since this leakage contributes less than 0.1 percent of the overall risk based on the existing regulations. The use of Option B will allow the extension of testing intervals with a minimal impact on the radiological release rates since most penetration leakage is continually well below the specified limits. In the accident risk evaluation, the NRC Staff noted that the accident risk is relatively insensitive to the containment leakage rate because the accident risk is dominated by accident sequences that result in failure of or bypass of the containment. The use of a performance-based testing program will continue to provide assurance that the accident analysis assumptions remain bounding. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

Removal of the surveillance accuracy requirement in Section 4.6.1.2.c will not affect the probability of an accident previously analyzed since a similar requirement is contained in ANSI/ANS-56.8-1994, ''Containment System Leakage Testing Requirements.'' ANSI/ANS-56.8-1994 will be used to develop the technical methods and techniques for the Containment Leakage Rate Test Program as stated in Regulatory Guide 1.163. The technical methods and techniques in ANSI/ANS-56.8-1994 have been determined to be acceptable to the NRC Staff.

Changes to the Administrative Section describe the containment testing program only and cannot increase the probability or consequences of an accident previously analyzed.

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

The proposed license amendment does not change the operation or equipment of the plant. The change in the test frequency is dependent on the establishment of a Containment Leakage Test Program. This test program will ensure the performance history of each penetration is satisfactory prior to the changing of any test frequency. Since the performance history of the penetration will be known, there is no possibility of the implementation of the program creating a new or different kind of accident than previously analyzed. Since there is no change to the equipment or the operation of the plant, there is no possibility of creating a new or different kind of accident than previously analyzed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

Removal of the surveillance accuracy requirement in Section 4.6.1.2.c will not create the possibility of a new or different kind of accident from those previously analyzed since a similar requirement is contained in ANSI/ANS-56.8-1994, ''Containment System Leakage Testing Requirements.'' ANSI/ANS-56.8-1994 will be used to develop the technical methods and techniques for the Containment Leakage Rate Test Program as stated in Regulatory Guide 1.163. The technical methods and techniques in ANSI/ANS-56.8-1994 have been determined to be acceptable to the NRC staff.

Changes to the Administrative Section describe the containment testing program only and cannot create a different accident from any previously analyzed.

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

During the development of 10 CFR Part 50, Appendix J, Option B, the NRC staff determined the reduction in safety associated with the implementation of the performance-based testing program. The results of this review are documented in NUREG-1493. The review concluded that reducing the frequency of Type A tests (Integrated Leakage Rate Tests) from three per 10 years to one per 10 years leads to an imperceptible increase in risk. For Type B and C testing (Local Leakage Rate Tests), the increase in testing frequency should not have significant impact since this leakage contributes less than 0.1 percent of the overall risk based on the existing regulations. The use of Option B will allow the extension of testing intervals with a minimal impact on the radiological release rates since most penetration leakage is continually well below the specified limits. In the accident risk evaluation, the NRC Staff noted that the accident risk is relatively insensitive to the containment leakage rate because the accident risk is dominated by accident sequences that result in failure of or bypass of the containment. The use of a performance based testing program will continue to provide assurance that the accident analysis assumptions remain bounding. Therefore, this change does not involve a significant reduction in the margin of safety.

Removal of the surveillance accuracy requirement in Section 4.6.1.2.c will not involve a significant reduction in the margin of safety since a similar requirement is contained in ANSI/ANS-56.8-1994, ''Containment System Leakage Testing Requirements.'' ANSI/ANS-56.8-1994 will be used to develop the technical methods and techniques for the Containment Leakage Rate Test Program as stated in Regulatory Guide 1.163. The technical methods and techniques in ANSI/ANS-56.8-1994 have been determined to be acceptable to the NRC Staff.

Changes to the Administrative Section describe the containment testing program only and do not reduce the margin of safety.

Moreover, the Commission has provided guidance concerning the application of standards in 10 CFR 50.92 by providing certain examples (51 FR 7751, March 6, 1986) of amendments that are considered not likely to involve an SHC [significant hazards consideration]. Although the proposed change is not enveloped by a specific example, it has been shown that the proposed change is not 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: Russell Library, 123 Broad Street, Middletown, CT 06457.

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

NRC Project Director: Phillip F. McKee

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: February 6, 1996

Description of amendment request: The proposed amendment would delete the requirement to perform additional operability testing of safety system train components when a required component in the redundant train becomes inoperable.

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

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

The proposed changes remove the requirement for testing which is in addition to the normal surveillance interval. The affected equipment is subject to periodic surveillance testing required by the Technical Specifications. Removing the requirement for additional testing cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing an AOT [allowable outage time] or a surveillance interval cannot increase the probability or consequences of an accident previously evaluated.

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

The proposed changes remove the requirement for testing which is in addition to the normal surveillance interval. The affected equipment is subject to periodic surveillance testing required by the Technical Specifications. Removing the requirement for additional testing cannot alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, changing an AOT or a surveillance interval cannot create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes remove the requirement for testing which is in addition to the normal surveillance interval, in effect extending the surveillance interval. An excessive surveillance interval extension could reduce the margin of safety by reducing assurance that required equipment will function as designed; an overly restrictive surveillance interval could also reduce the margin of safety by imposing unnecessary testing wear, equipment manipulations, and system transients on the plant.

The existing requirements to perform cross-train testing were based on the operating experience available when they were added to the TS. Typically this was done during the initial plant licensing in 1971. The recently published Standard Technical Specifications (NUREG 1432) do not include cross-train testing requirements for the Engineered Safety Features components. It has been judged by the NRC and by the industry, that cross-train testing is unnecessary, and that testing at normal surveillance intervals is adequate to assure equipment operability. This recent judgment is based on a much larger accumulation of operating experience than was available at the time Palisades was licensed. There are no special features of the Palisades plant which would invalidate these more recent judgments of optimal testing requirements. Therefore, operation of the facility in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Van Wylen Library, Hope College, Holland, Michigan 49423

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

NRC Project Director: Mark Reinhart

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

Date of amendment request: December 14, 1995, as supplemented by letter dated May 16, 1996.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) to improve the TS Action Statements and Surveillance Requirements for diesel generators in accordance with the recommendations and guidance in Generic Letter 93-05, Generic Letter 94-01, NUREG-1366, and NUREG-1431. The proposed amendments would also incorporate technical and administrative 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:

Criterion 1

Operation of the facilities in accordance with the requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Improvements to the LCOs [limiting condition for operation] and surveillance requirements for the emergency diesel generators do not affect their capability to provide emergency power to plant vital instruments and safety related equipment. In fact, these improvements make the diesel generators more reliable since they significantly reduce the amount of wear and stress due to excessive and unnecessary testing. The proposed monthly testing of the diesel generator continues to ensure that the system is ready for service when needed. The fast starts and fast loadings continue to ensure that the timing and loading requirements for engineered safety features actuation are met. The proposed changes do not affect any of the design basis accident analyses previously evaluated. Therefore, these proposed changes do not involve any increase in the probability or consequences of any accident previously evaluated. The proposed changes are fully consistent with the recommendations and guidance contained in GL [Generic Letter] 93-05, GL 94-01, NUREG-1366, NUREG-1431, and are compatible with plant operating experience.

Criterion 2

Operation of the facilities in accordance with the requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes in fact improve the reliability of the diesel generators by eliminating unnecessary wear and stress. Improved reliability decreases the failure probability which also decreases the probability of an accident not previously evaluated. None of the requested amendments increase the common mode failure probability thus would not increase the chance of both EDG's [emergency diesel generators] for a particular nuclear unit being out of service simultaneously. The proposed changes are fully consistent with the recommendations and guidance contained in GL 93-05, GL 94-01, NUREG-1366, NUREG-1431, and are compatible with plant operating experience.

Criterion 3

Operation of the facilities in accordance with the requested amendments will not involve a significant reduction in a margin of safety. The proposed monthly testing of the diesel generators continues to ensure that the system is ready for service when needed. The fast starts and fast loadings continue to ensure that the timing and loading requirements for engineered safety features actuation are met. The proposed changes improve the reliability of the diesel generators. Implementation of the Maintenance Rule also ensures continued reliability of the diesel generators. No margin of safety is decreased as a result of these TS changes.

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: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

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

NRC Project Director: Herbert N. Berkow

Entergy Operations, Inc., Docket No. 50-313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: April 29, 1996

Description of amendment request: The proposed amendment relocates several cycle specific operating parameters from the technical specifications to the Core Operating Limits Report per Generic Letter 88-16. The parameters being relocated by this change include the variable low reactor coolant system pressure trip (VLPT) and the variable low reactor coolant system pressure-temperature protective limits.

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

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

The removal of the cycle-dependent variable low RCS pressure-temperature protective limits and the VLPT setpoint from technical speciications and placing them into the COLR has no impact on plant safety and is considered to be administrative in nature. The proposed change does not affect the safety analyses, physical design, or operation of the plant. Technical specifications will continue to require operation within the core protective and operational limits for each reload cycle as calculated by the approved reload design methodologies. The appropriate actions required if limits are violated will remain in the technical specifications. The reload report presents the results of cycle-specific evaluations of accident analyses and transients addressed in the ANO-1 Safety Analysis Report. The cycle-specific 10CFR50.59 evaluation of the reload report demonstrates that changes in fuel cycle design and the corresponding COLR do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

The proposed change to relocate the variable low RCS pressure-temperature protective limits and the VLPT setpoint from the technical specifications to the COLR is administrative in nature. No change to the design configuration or method of operation of the plant is made by this proposed change, and therefore, no new transient initiator has been created. Technical specifications will continue to require operation within the required core protective and operating limits and appropriate actions will be taken if the limits are exceeded. Because plant operation will continue to be limited by the cycle-specific COLR limits that are established using NRC-approved methodologies, these relocations will have no impact on plant safety.

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

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

Existing technical specification operability and surveillance requirements are not reduced by the proposed change to relocate the variable low RCS pressure-temperature protective limits and the VLPT setpoint to the COLR. The proposed changes are administrative in nature and do not relate to or modify the safety margins defined in and maintained by the technical specifications. The cycle-specific COLR limits for future reload fuel cycles will continue to be developed based on NRC approved methodologies. Each future reload undergoes a 10CFR50.59 evaluation to assure that operation of the plant within the cycle-specific limits will not involve a significant reduction in a 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., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 6, 1996

Description of amendment request: The amendment would reflect that the name of Mississippi Power & Light Company (MP&L) has been changed to Entergy Mississippi, Inc. The amendment revises Operating License NPF-29 and Antitrust Conditions for the Grand Gulf Nuclear Station, Unit 1 (GGNS) to (1) add the phrase ''(now renamed Entergy Mississippi, Inc.)'' after the name of Mississippi Power & Light Company (MP&L), (2) replace the name of Mississippi Power & Light Company (MP&L) by the name Entergy Mississippi, Inc., and (3) replace a footnote by the statement: ''Amendment —— resulted in a name change for Mississippi Power & Light Company (MP&L) to Entergy Mississippi, Inc.''.The proposed amendment involves only a change in company name. It does not involve any changes to the Technical Specifications for GGNS, or to any requirements or limiting conditions for operation on any equipment or any systems in the plant.

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:

Entergy Operations, Inc. proposes to change the current Grand Gulf Nuclear Station Facility Operating License and Antitrust Conditions. The specific proposed change is to reflect that the name of one of the companies owning Grand Gulf Nuclear Station has legally changed from Mississippi Power & Light Company to Entergy Mississippi, Inc.

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if 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.

Entergy Operations, Inc. has evaluated the no significant hazards consideration in its request for this license amendment and determined that no significant hazards consideration results from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change documents changing the legal name of the company. The proposed change will not affect any other obligations. The company will still own all of the same assets, serve the same customers, and all existing obligations and commitments will continue unaffected.

[The proposed change does not affect any of the existing requirements or commitments on equipment or systems that are designed for the safe operation of the plant. It does not affect the design or operation of the plant.]

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

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

The administrative changes to the Operating License [and Antitrust Condition] requirements [to change the name of Mississippi Power & Light] do not involve any change in the design or operation of the plant. The company will still own all of the same assets, serve the same customers, and all existing obligations and commitments will continue unaffected.

[The proposed changes do not affect equipment or systems that could caused an accident at the plant.]

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

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

The proposed change [in name] is administrative in nature, as described above; therefore, this change does not reduce the level of safety imposed by any current requirements. [The proposed changes do not affect any equipment or systems at the plant.] The company will still own all of the same assets, serve the same customers, and all existing obligations and commitments will continue unaffected.

Therefore, the proposed changes do not cause 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. herefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 8, 1996

Description of amendment request: The amendment request would replace the current frequency requirements in Surveillance Requirement (SR) 3.6.1.3.5, on the leakage rate testing for each containment purge valve with resilient seals, in the Technical Specifications for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed changes would place these purge valves on a performance-based leakage testing frequency, instead of the current once every 184 days and once within 92 days after opening the valve.The proposed changes do not change the limiting conditions for operation, the required actions for inoperability, or the other surveillance requirements on these primary containment isolation valves.

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

In accordance with 10 CFR 50.92, Entergy Operations, Inc. has evaluated the proposed change to the Operating License of GGNS and has determined that the operation of the facility in accordance with the proposed amendment would not involve any significant hazards considerations. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the following analysis of the proposed amendment against the three [following] standards of 10 CFR 50.92(c):

1) The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

This change deletes the augmented testing requirement for these containment isolation valves and allows the surveillance intervals to be set in accordance with the Appendix J testing program. [Appendix J to 10 CFR Part 50 defines primary containment leakage testing requirements for water-cooled power reactors as GGNS and these requirements include frequency of testing for the primary containment isolation valves.] This change does not affect the system function or design. The purge valves are not an initiator of any previously analyzed accident. Leakage rates do not affect the probability of the occurrence of any accident. Operating history has demonstrated that these valves do not degrade and cause leakage as previously anticipated. Because these valves have been demonstrated to be reliable, these valves can be expected to perform the containment isolation function as assumed in the accident analyses.

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

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

Extending the test intervals has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. No change has been made to the design, function or method of performing leakage testing [or to the design and function of these valves]. Leakage acceptance criteria have not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change.

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

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

The only margin of safety that has the potential of being impacted by the proposed changes involves the offsite dose consequences of postulated accidents which are directly related to the containment leakage rate. The proposed change does not alter the method of performing the tests nor does it change the leakage acceptance criteria. Sufficient data has been collected to demonstrate that the resilient seals do not degrade at an accelerated rate.

[Also, the proposed change would test these valves in accordance with the Appendix J testing program at the plant. Appendix J to 10 CFR Part 50 defines primary containment leakage testing requirements for water-cooled power reactors as GGNS and these requirements include frequency of testing for the primary containment isolation valves.]

Because of this demonstrated reliability, this change will provide sufficient surveillance to determine an increase in the unfiltered leakage prior to the leakage exceeding that assumed in the accident analysis.

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

Based on the above evaluation, Entergy Operation, Inc. has concluded that operation in accordance with the proposed amendment involves no significant hazards considerations.

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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 9, 1996

Description of amendment request: The amendment request would (1) increase the safety limit minimum critical power ratio (MCPR) for two loop operation and single loop operation to 1.10 and 1.11, respectively, and (2) add a General Electric topical report to the list of documents describing the analytical methods used to determine the core operating limits. The proposed changes are to Section 2.1.1, Reactor Core Safety Limits, and Section 5.6.5, Core Operating Limits Report (COLR), respectively, of the Technical Specifications (TSs).

The licensee also proposed changes to the Bases of the TSs associated with the above proposed changes.

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

Entergy Operations, Inc. proposes to change the current Grand Gulf Nuclear Station [GGNS] Technical Specifications. The specific change is to modify the Minimum Critical Power Ratio (MCPR) safety limits reported in Technical Specification 2.1.1.2, the list of references in Technical Specification 5.6.5, and associated Bases changes. The proposed change is necessary in order to switch reload fuel vendors. [General Electric GE11 fuel is being added to the core in place of Siemens Power Corporation (SPC) fuel.]

The Commission has provided standards for determining whether no significant hazards considerations exists as stated in 10 CFR 50.92 (c). A proposed amendment to an operating license involves no significant hazards if 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; (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.

Entergy Operations, Inc. has evaluated the no significant hazards consideration in its request for this license amendment and determined that no significant hazards considerations result from this change. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification 2.1.1 as that limit which ''ensures that during normal operation and during Anticipated Operational Occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling.'' The MCPR safety limit is re-evaluated for each reload and, for GGNS [Operating] Cycle 9, the analyses have concluded that a two-loop MCPR safety limit of 1.10 based on the application of the generic GE MCPR methodology is necessary to ensure that this acceptance criterion is satisfied. For single-loop operation, a MCPR safety limit of 1.11 based on the generic GE MCPR methodology was determined to be necessary. Core MCPR operating limits are developed to support the Technical Specification 3.2 requirements and ensure these safety limits are maintained in the event of the worst-case transient. Since the MCPR safety limit will be maintained at all times, operation under the proposed changes will ensure at least 99.9% of the fuel rods in the core do not experience transition boiling. Therefore, The Minimum Critical Power Ratio (MCPR) safety limit change does not affect the probability or consequences of an accident.

The implementation of GE's GESTAR-II approved methodology has no effect on the probability or consequences of any accidents previously evaluated. One exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current licensing basis. The design of the GE11 fuel bundles is such that the bundles are not likely to be mis-oriented or mis-located and the normal administrative controls will be in effect for assuring proper orientation and location. Therefore, the probability of a fuel loading error is not increased. This analysis ensures that postulated dose releases will not exceed a small fraction (10 percent) of 10CFR100 limits.

Therefore, the consequences of accidents previously evaluated are unchanged.

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

The GE11 fuel to be used in [Operating] Cycle 9 is of a design compatible with fuel present in the core and used in the previous cycle. Therefore, the GE11 fuel will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. They introduce revised MCPR safety limits that have been proved to be acceptable for Cycle 9 operation. Compliance with the applicable criterion for incipient boiling transition continues to be ensured. The proposed MCPR safety limits do not result in the creation of any new precursors to an accident.

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

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

The MCPR safety limits have been evaluated to ensure that during normal operation and during AOOs [abnormal operating occurrences], at least 99.9% of the fuel rods in the core do not experience transition boiling. Therefore, the implementation of the proposed changes in the MCPR safety limit ensure there is no reduction in the margin of safety.

As with the current SPC methodology, GGNS will implement only the NRC-approved revisions to GE's GESTAR methodology. This GE methodology is similar to those SPC reports currently listed in TS 5.6.5 and it will be applied in a similar, conservative fashion. One exception to GESTAR is that the mis-oriented and mis-located bundle events will continue to be analyzed as accidents subject to the acceptance criteria in the current licensing basis. This analysis ensures that postulated dose releases will not exceed a small fraction (10 percent) of 10CFR100 limits. On this basis, the implementation of this GE methodology 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: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, 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: February 6, 1996

Description of amendment request: The proposed change will amend the Allowable Values of parameters in Table 3.3-4 of Waterford Steam Electric Station, Unit 3, (Waterford 3) Technical Specifications (TSs) to make it consistent with the identical parameters in Table 2.2-1 of TSs for Waterford 3. The proposed change will add Mode 4 to the surveillance requirements of Table 4.3-2, Item 5.c (Safety Injection System Automatic Actuation Logic) that was inadvertently removed. Finally, the proposed change removes a reference to TS 3.3.3.2 in Surveillance Requirements TS 4.10.2.2 and 4.10.4.2 since Incore Detectors has been removed from the TSs.

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

The proposed changes described herein are administrative changes necessary to correct administrative errors. The proposed changes will have no affect on any design basis accidents nor will these changes affect any material condition of the plant. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed changes are purely administrative. There are no new system or design changes associated with this proposal. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will have no impact on any protective boundary, safety limit, or margin to safety. The proposed change corrects inconsistencies in the TS and is purely administrative in nature. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: William D. Beckner

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: May 7, 1996 (TSCR 247)

Description of amendment request: The proposed change to the technical specifications would adopt the provisions of the Standard Technical Specifications (STS), NUREG-1433, Rev. 1, which clarify surveillance requirement applicability and allow a maximum period of 24 hours to complete a surveillance requirement upon discovery that the surveillance has been missed.

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

1. Operation of the facility in accordance with the proposed amendment would not

involve a significant increase in the probability of occurrence or consequence of an accident previously evaluated. The proposed changes only affect administrative requirements regarding the applicability and performance of surveillances. This change clarifies surveillance requirement applicability and allows a maximum 24 hour delay period for the performance of a surveillance when it is discovered that the surveillance has not been performed within the required frequency, consistent with the STS. There is minimal safety significance associated with a delay of 24 hours in completing the required surveillance, particularly due to the fact that the most probable result of any particular surveillance performed is the successful verification of conformance with the requirements.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes only affect administrative requirements regarding the applicability of surveillance requirements and the performance of surveillances to allow a maximum 24 hour delay period when it is discovered that a surveillance has been missed. No changes to plant equipment or operation are affected.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety since the change contained in the proposed amendment does not change any existing safety margins.

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: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

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

NRC Project Director: John F. Stolz

GPU Nuclear Corporation, Docket No. 50-320, Three Mile Island Nuclear Station, Unit No. 2 (TMI-2), Dauphin County, Pennsylvania

Date of amendment request: February 16, 1995

Description of amendment request: The proposed amendment would revise TMI-2 Operating License No. DPR-73 by modifying sections 4.02, 4.04, and 4.1.1.3 of the unit technical specifications. The revisions to sections 4.02 and 4.04 would add flexibility to the scheduling of surveillance activities and would allow for a 24 hour period to perform missed surveillances before declaration of a limiting condition for operation, respectively. These changes would make the TMI-2 technical specifications consistent with the Standard Technical Specifications for B&W Plants (NUREG-1430). The revision to section 4.1.1.3 would allow extension of the time interval for surveillance of the containment airlock doors from quarterly to annually. The proposed changes to the TMI-2 technical specifications section 4.1.1.3 would allow a decrease in worker exposure to radiation while maintaining an adequate level of environmental protection at the facility.

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:

10 CFR 50.92 provides the criteria which the Commission uses to perform a no significant hazards consideration. 10 CFR 50.92 states that an amendment to a facility license involves no significant hazards if 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 proposed changes to the technical specifications sections 4.02 and 4.04 are administrative and do not involve any physical changes to the facility. No changes are made to operating limits or parameters, nor to any surveillance activities. The changes to section 4.1.1.3 extends the interval between surveillance of the containment airlocks; it does not change the operability requirements, test methodology or acceptance criteria. Based on this, GPU Nuclear has concluded that the proposed changes to sections 4.02 and 4.04 do not:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The changes do not modify any operating parameters or the release of radioactive materials. The clarification of maximum time extensions for surveillance is consistent with the NRC's Standard Technical Specifications for Babcock and Wilcox Plants (NUREG-1430).

2. Create the possibility of a new or different kind of accident since these change are administrative and no plant configuration or operational changes are involved.

3. Involve a change in the margin of safety. These changes are administrative in nature, compatible with standard technical specifications, and do not affect any safety settings or operational limits.

GPU Nuclear has also concluded that the proposed changes to section 4.1.1.3 do not:

1. Involve a significant increase in the probability of occurrence of or consequences of an accident previously evaluated. The change to this section does not change operating parameters, equipment operability requirements, surveillance test methodology, or acceptance criteria.

2. Create the possibility of a new or different kind of accident since the change does not affect plant equipment, plant configuration, or plant operating parameters.

3. Involve a change in the margin of safety since the change does not affect any operational limits.

Based on the above analysis the licensee concluded that the proposed changes involve no significant safety hazards considerations as defined by 10 CFR 50.92.

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

Local Public Document Room Location: Government Publications Section, State Library of Pennsylvania, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105

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

NRC Project Director: Seymour H. Weiss

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

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendments would change the Technical Specifications to implement 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leakage-Test Program.''

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:

[South Texas Project] STP has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of STP in accordance with the proposed amendment will not:

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

10 CFR [Part] 50, Appendix J has been amended to include provisions regarding

performance based leakage testing requirements (Option B). Option B allows plants with satisfactory Integrated Leak Rate Testing (ILRT) performance history to extend the Type A testing interval from three tests in ten years to one test in ten years. For Type B and Type C tests, Option B allows extended testing interval[s] based on the leak rate test history of each component. To be consistent with the requirements of 10 CFR [Part] 50, Appendix J, Option B, STP proposes to include appropriate changes to the Technical Specifications that incorporate the necessary revisions associated with 10 CFR [Part] 50, Appendix J, Option B.

The proposed amendment represents the conversion of current Technical Specification requirements to maintain consistency with those requirements specified by 10 CFR [Part] 50, Appendix J, Option B. The proposed changes are consistent with the current safety analyses. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits, and will not significantly increase the probability or consequences of a previously evaluated accident.

Some proposed changes represent minor relaxations in current Technical Specification requirements, but are based on the requirements specified by Option B of 10 CFR [Part] 50, Appendix J. Changes are consistent with the current safety analyses and determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analyses, and provide continued assurance that specified parameters associated with containment integrity remain within their acceptance limits. These changes will not significantly increase the probability or consequences of a previously evaluated accident.

The systems affecting containment integrity related to this proposed amendment request are not assumed in any safety analyses to initiate any accident sequence. The probability of any accident previously evaluated is not increased by this proposed amendment. The proposed changes to Technical Specification LCOs or SRs maintain an equivalent level of reliability and availability for all affected systems. The proposed amendment does not increase the consequences of any accident previously evaluated.

There is no change to the consequences of an accident previously evaluated because maintaining leakage within the analyzed limit assumed for any associated accident analyses does not adversely affect either the on-site or off-site dose consequences resulting from an accident. There is no adverse impact on the probability of accident initiators. There is no significant increase in the probability of any previously analyzed accident. A plant specific risk-based analysis of Appendix J performed for STP indicates the containment penetration leakage dose rate contribution to the total dose rate in person-rem is insignificant.

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

10 CFR [Part] 50, Appendix J, Option B specifies, in part, that a Type A test which measures both the containment system overall integrated leakage rate at containment pressure and system alignments assumed during a large break LOCA [loss-of-coolant accident], and demonstrates the capability of primary containment to withstand an internal pressure load, may be conducted at an interval based on the performance of the overall containment system. The acceptable leakage rates are specified in the plant's Technical Specifications. For Type B and Type C tests, intervals are proposed based on the performance history of each component. Acceptance criteria for each component is based upon demonstration that the sum leakage rates at design basis pressure conditions for applicable penetrations, is within the limit specified in the Technical Specifications.

The proposed amendment represents the conversion of current Technical Specification requirements to maintain consistency with those requirements specified in 10 CFR [Part] 50, Appendix J, Option B. The proposed changes are consistent with the current safety analyses. Some minor relaxations in current Technical Specification requirements, associated with containment integrity are based on generic guidance provided in Option B, NEI 94-01 and ANSI/ANS 56.8, 1994. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the testing of components; however, these are in accordance with the STP current safety analyses and provide for appropriate testing or surveillance that are consistent with 10 CFR [Part] 50, Appendix J, Option B. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment has been reviewed for acceptability considering similarity of system or component design affecting containment integrity. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect corresponding changes associated with Option B of 10 CFR [Part] 50, Appendix J and improvements in technique or interval of leak rate testing performance. The proposed changes maintain, at minimum, the present level of operability of any system that affects containment integrity. The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that affect leak rate integrity related to the proposed amendment, are not assumed in any safety analysis to initiate any accident sequence. The proposed surveillance requirements for any affected systems are consistent with the current requirements specified within the Technical Specifications and are consistent with the requirements of Option B of 10 CFR [Part] 50, Appendix J. The proposed surveillance requirements maintain an equivalent level of reliability and availability of all affected systems and therefore, does not increase the consequences of any previously evaluated accident.

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

The provisions specified in Option B of 10 CFR [Part] 50 Appendix J allow changes to Type A, Type B, and Type C test intervals based upon the performance of past leak rate tests. The effect of extending containment leakage rate testing intervals has a corresponding increase in the likelihood of containment leakage. The degree to which intervals can be extended is a direct function of the potential effect to existing safety margins and the public health and safety that can occur due to an increased likelihood of containment leakage. 10 CFR [Part] 50 Appendix J, Option B allows longer intervals between leakage tests based on performance trends but does not increase the leakage acceptance criteria. La [maximum allowable leakage rate] is still limited to 0.3 wt%/day. By referencing the Containment Leakage Rate Testing Program in LCO 3.6.1.2 ACTION, the point at which ACTION is required is increased from .75 La to 1.0 La. This makes the specification consistent with the intent of having margin between an AS-LEFT leakage of less than or equal to .75 La and maintaining operability with less than or equal to 1.0 La.

Changing Appendix J test intervals from those currently provided in the Technical Specification to those provided in 10 CFR [Part] 50, Appendix J, Option B, slightly increases the risk associated with Type A, Type B, and Type C specified accident sequences. Historical data suggests that increasing the Type C test interval can slightly increase the associated risk; however, this is compensated by the corresponding risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. When considering the total integrated risk which includes all analyzed accident sequences, the risk associated with increasing test intervals is negligible. A plant specific risk-based analysis of Appendix J performed for STP indicates the containment penetration leakage dose rate contribution to total dose rate in person-rem is insignificant.

STP proposes to revise the Technical Specifications to be consistent with those provisions specified in Option B of 10 CFR, Appendix J. The proposed changes are consistent with the STP current safety analyses. These proposed changes do not involve revisions to the design of the station. The proposed changes will maintain the same level of reliability of equipment associated with containment integrity assumed to operate in the safety analysis, and provide continued assurance that specified parameters affecting plant leak rate integrity will remain within acceptance limits. The proposed changes provide continued assurance of leakage integrity of containment without adversely affecting the public health and safety and will not significantly reduce existing safety margins. Plant specific risk-based analysis indicates sufficient technical justification exists to further extend the limits beyond those allowed by Option B.

The proposed amendment to the Technical Specifications implements present requirements, or the requirements in accordance with the guidelines set forth in Option B of 10 CFR [Part] 50, Appendix J. NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' served as the technical basis for Option B. STP performed a plant specific risk-based analysis of containment penetration leakage dose utilizing the same methodology used in NUREG-1493. The analysis indicates the containment penetration leakage dose rate contribution to the total dose rate in person-rem is insignificant. This plant specific analysis serves to validate the applicability of the proposed changes for STP. The proposed changes have been approved by the NRC, are applicable to STP, maintain necessary levels of system or component reliability affecting containment integrity, and do not involve a significant reduction in the margin of safety.

The performance-based approach to leakage rate testing concludes the impact on public health and safety due to revised testing intervals is negligible. The proposed amendment will not reduce availability of systems associated with containment integrity when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Guidance has been provided in ''Final Procedures and Standards on No Significant Hazards Considerations,'' Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for limiting safety system settings or a significant relaxation of the bases for LCOs. Therefore, based on the guidance provided in the Federal Register and criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

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

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

Attorney for licensee: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036-5869

NRC Project Director: William D. Beckner

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

Date of amendment requests: December 14, 1995

Description of amendment requests: The proposed amendments would revise the Administrative Control (Chapter 6) Section and other affected Sections of the Prairie Island Technical Specifications to generally conform with NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Revision 1, dated April 7, 1995.

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

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

Operation of the Prairie Island plant in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. None of the proposed changes involve a physical modification to the plant, a new mode of operation or a change to the Updated Safety Analysis Report transient analyses. These proposed amendments generally conform to the guidance of NUREG-1431, Revision 1, Section 5.0 which was previously reviewed, accepted and issued by the NRC.

Some Section 5.0 Specifications in NUREG-1431 were not incorporated in this License Amendment Request. These Specifications were not proposed because they 1) specify requirements not currently in the Prairie Island Technical Specifications or otherwise committed to, 2) are addressed elsewhere in the current Technical Specifications, or 3) the current Technical Specifications level of commitment is maintained. In all these instances, the NRC has previously reviewed and approved the proposed level of commitment through the issuance of the current Prairie Island Technical specifications.

The proposed changes, in themselves, do not reduce the level of qualification or training such that personnel requirements would be decreased.

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

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

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes, in themselves, do not introduce a new mode of plant operation, surveillance requirement or involve a physical modification to the plant. These proposed amendments generally conform to the guidance of NUREG-1431, Revision 1, Section 5.0 which was previously reviewed, accepted and issued by the NRC.

Some Section 5.0 Specifications in NUREG-1431 were not incorporated in this License Amendment Request. These Specifications were not proposed because they 1) specify requirements not currently in the Prairie Island Technical Specifications or otherwise committed to, or 2) are addressed elsewhere in the current Technical Specifications. Other features are not fully implemented but rather, the current Technical Specification level of commitment is maintained. In all these instances, the NRC has previously reviewed and approved the proposed level of commitment through the issuance of the current Prairie Island Technical Specifications.

In general, the proposed changes are administrative in nature. The changes propose to revise, delete or relocate Specifications within the Technical Specifications or from the Technical Specifications to the Updated Safety Analysis Report, plant procedures or the Operational Quality Assurance Plan through which adequate control is maintained. The proposed changes do not alter the design, function, or operation of any plant components and therefore, no new accident scenarios are created.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated would not be created [by] these amendments.

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

The proposed changes do not involve a significant reduction in a margin of safety because the Current Technical Specifications requirements for safe operation of the Prairie Island plant are maintained or increased. The proposed changes are administrative in nature and do not involve a physical modification to the plant, a new mode of operation or a change to the Updated Safety Analysis Report transient analyses. The proposed changes do not alter the scope of equipment currently required to be operable or subject to surveillance testing nor does the proposed change affect any instrument setpoints or equipment safety functions.

Therefore, a significant reduction in the margin of safety would not be involved with these amendments.

Based on the evaluation describe above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation [of] the Prairie Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by Nuclear Regulatory Commission regulations in 10 CFR Part 50, Section 50.92.

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

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

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

NRC Project Director: Mark Reinhart (Acting Director)

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

Date of amendment requests: February 15, 1996

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 to revise Technical Specification 3.5.2, ''ECCS Subsystems - Tavg Greater Than or Equal to 350°F,'' to change the allowed outage time for any one safety injection pump from 72 hours to 7 days. The specific TS change proposes to add a new footnote that increases the allowed outage time (AOT) for one safety injection (SI) pump from 72 hours to 7 days for performance of non-routine, emergent maintenance and requires review by the Plant Staff Review Committee (PSRC), and requires Plant Manager approval prior to exceeding 72 hours.

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

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

The proposed allowed outage time (AOT) extension does not change the operating practices of Diablo Canyon Power Plant (DCPP). Although the proposed change increases the allowed time in which the safety injection (SI) system may be out of service for maintenance or testing, this extended AOT will only be used in emergent circumstances.

Increasing the AOT for the SI pumps does not involve physical alteration of any plant equipment and does not affect analysis assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents. Further, the severity of postulated accidents and resulting radiological effluent releases will not be affected by the increased AOT.

Finally, the probabilistic risk assessment determined that the increase in the core damage probability is not considered significant.

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

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

The proposed increase to the SI pump AOTs does not change the method by which DCPP operates. Further, the proposed change would not result in any physical alteration to any plant system, and there would not be a change in the method by which any safety related system performs its function.

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

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

There is no safety analysis impact since the extension of the SI pump AOT interval will have no effect on any safety limit, protection system setpoint, or limiting condition of operation. There is no hardware change that would impact existing safety analysis acceptance criteria.

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

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

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

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

NRC Project Director: William H. Bateman

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

]Date of application request: April 17, 1996

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3/4.3 to support a future modification to replace existing digital portions of the main steam and feedwater isolation system (MSFIS) with digital processor equipment and would authorize revision of the FSAR to include a description of the MSFIS modification. The MSFIS modification is a change to the facility, as described in the safety analysis report, that involves an unreviewed safety question. The modification involves an unreviewed safety questions because: (1) the MSFIS design will use software which could result in a common mode failure, (2) the original NRC review of the MSFIS did not evaluate 2 out of 3 coincidence circuitry, which could introduce new system failure modes, and (3) the MSFIS modification utilizes manual handswitches that could introduce new system failure modes. The NRC will review the modification in accordance with 10 CFR 50.59(a)(2) in conjunction with the review of the proposed TS amendment.

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 addition of the MSFIS actuation logic and relays to the TS has no adverse impact on the probability of occurrences or the consequences of an accident. The proposed amendment does not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident and the methodologies used in the accident analysis remain unchanged. The operating limits will not be changed.

No design basis accidents will be affected by this design change since the logic which currently exists will continue to be performed. Thus, the radiological consequences will not change.

The system response time is enveloped by the current 5 second valve stroke time. The MSFIS response time will be less than 500 msec.

A common mode software failure could exist if both separation groups have their PLCs [programmable logic controllers] (3 per train - six total) malfunction at the same time. However, a diverse means of isolating the feedwater lines exists given the ability of the Main Feed Control Valves to close on a Feedwater Isolation Signal. The MSIVs [main steam isolation valves] do not have a diverse means of isolating their respective steam lines if a common mode software failure occurs. As a result, this modification provides a means to manually fast close the valves at the MSFIS cabinets. The operators will be alerted of the failure conditions of any PLC logic channel via MCB [main control board] annunciators and indicators. This failure mode has a low probability of occurrence based upon the inherent quality of the design provided by the V&V [verification & validation] process. Therefore, the accident consequences are not increased for this failure mode.

The test panel in the MSFIS cabinets has been laid out to provide the same functions as the existing test panel, except that PLC status indication and coincidence logic test functions are provided. The Emergency Override Panel, located below the Test Panel, provides the operator with the ability to bypass the FWIS [feedwater isolation signal] signal and manually fast close each MSIV as required by the Emergency Operating Procedures. The MSIV manual FC [fast close] switch operation is necessary for a diverse means of operation for software common mode failures. The FWIS bypass switch will allow main feedwater flow to be re-established to each Steam Generator.

The replacement system is functionally the same as the current system since it performs the same logic, receives the same inputs, and produces the same outputs. However, the system is more reliable and possesses triple redundant logic. Therefore, the probability of malfunction will not be increased.

The electrical load of the A-B PLC equipment and existing 48 VDC [volts direct current] actuation relays is less than that of the existing equipment so the system will not require any additional cooling over the existing equipment. Proper grounding is provided for the PLC 5 VDC and actuation relay 48 VDC power supplies, which are electrically isolated from each other.

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

The addition of the MSFIS actuation logic and relays to the TS will not create a new type of accident or malfunction than any previously evaluated in the Safety Analysis Report. The safety functions of the system are not changed in any manner, nor is the reliability of any structure, system or component reduced. All design and performance criteria continue to be met. Since the safety functions and reliability are not adversely affected, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operator's ability to adequately respond to an accident is not hindered by the man-machine interface added as a result of this modification since the operator interface is similar to the current system and the MCB controls will not change. The operators will be alerted to system malfunctions through annunciation. The current system has a status output for each MSIV and FIV [feedwater isolation valve] valve on the Engineered Safety Feature Status Panel, which will be maintained. In addition, an isolated plant annunciator interface will provide a MSFIS Channel Failure plant annunciator window for both trains. Training will be provided to the technicians, engineers, and operators on the new features of the system prior to installation. Therefore, this modification does not increase the consequential effects due to the man-machine interface.

The system is compatible with the normal and accident environments and will be seismically qualified in accordance with the SNUPPS [standardized nuclear unit power plant system] seismic spectra profile. The equipment will be qualified for Electromagnetic Interference concerns in accordance with EPRI [Electric Power Research Institute] document TR-102323-EPRI Guideline and will meet the EPRI EMI [electromagnetic interference] limiting practices.

The system has the same failure mode upon loss of power as the current system and behaves similarly upon power restoration. A loss of power will not result in a MSFIS actuation.

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

The addition of the MSFIS actuation logic and relays to the TS will not affect or change a safety limit or affect plant operations. This change will not reduce the margin of safety assumed in the accident analysis nor reduce any margin of safety as defined in the basis for any TS.

The system response time for any given valve will not exceed the required valve stroke time. Since the MSFIS does not contain any analog channels, no channel trip accuracies are impacted.

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

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: October 25, 1995

Description of amendment request: The proposed changes would provide an allowed outage time of 14 days for the pressurizer power-operated relief valve (PORV) nitrogen accumulators, as well as provide separate action statements for the PORV depending on the reason for the PORV inoperability during plant operation in power Modes 1, 2, or 3.

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.

Specifically, operation of North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

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

The PORVs are assumed to mitigate the consequences of a steam generator tube rupture as described in the North Anna UFSAR [Updated Final Safety Analysis Report] as well as to limit the undesired opening of the pressurizer safety valves for a primary overpressure event. The proposed action statements ensure that the steam generator tube rupture accident analysis requirements are met. The proposed Technical Specification changes require the backup nitrogen supply be available for the PORVs to be consideredoperable and add action statements and surveillance requirements for the nitrogen supply commensurate with its significance. The proposed action statements enhance the availability of the automatic actuation of the PORVs by not requiring the block valves to be closed when the backup nitrogen supplies are inoperable. The proposed surveillance requirements enhance the reliability of the backup nitrogen supply to the PORVs by verifying that there is sufficient nitrogen pressure in the accumulators for the PORVs to perform their design function. The proposed Technical Specification changes do not change any accident analyses, therefore, the probability of any accident and its resulting consequences are not increased.

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

The proposed Technical Specification changes do not involve any physical modification to the plant or result in a change in a method of operation. The backup nitrogen supply continues to be required for PORV operability. The proposed Technical Specification changes provide operational flexibility and ensure the availability of the PORVs using the normal supply of instrument air while the backup nitrogen supply is being restored. This also prevents undesirable challenges to the pressurizer safety valves. The new surveillance requirements verify that there is sufficient nitrogen pressure in the accumulators for the PORVs to perform their design functions.

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

The proposed Technical Specification changes do not affect any safety limits or limiting safety system settings. The availability of the PORVs will be maintained as required in Generic Letter 90-06. The proposed Technical Specifications will continue to ensure that the PORVs will be capable of performing their intended functions.

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

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

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

NRC Project Director: Eugene V. Imbro

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

Date of amendment request: April 24, 1996

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 15.7, ''Radiological Effluent Technical Specifications (RETS).'' Portions of the RETS would be moved to licensee-controlled documents consistent with Nuclear Regulatory Commission guidance on TS improvements. Changes to other sections of the TSs are also proposed consistent with the removal of portions of the RETS.

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?

The proposed amendment simplifies the RETS and implements the recommendations of GL 89-01 and of GL 95-10. The proposed change relocates the operational requirements of RETS but keeps the programmatic controls for these requirements in the Technical Specifications. Therefore, the proposed changes are administrative in nature and do not affect plant operations. Hence, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because no safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. Also, the changes are unrelated to the initiation and mitigation of accidents and equipment malfunctions addressed in the Final Safety Analysis Report.

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

As stated above, the proposed action is the relocation of the RETS procedural details to various manuals while retaining the administrative controls in RETS. The relocation is consistent with the intent of the guidance of GL 89-01 and of GL 95-10. It is administrative and has no impact on plant operation or safety. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. No changes to plant components or structures are introduced which could create new accidents or malfunctions not previously evaluated.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenario is created and no previously evaluated accident scenario is changed by the relocation of the procedural details of RETS from one controlled document to another.

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

The proposed change does not include a change to any plant structure, system, component, or operation. The proposed changes do not alter the basic regulatory requirements and do not affect any safety analyses. The proposed change is administrative. The procedural details of the current RETS are relocated while the programmatic controls consistent with regulatory requirements, including controls on revisions to the manuals receiving the RETS procedural details, the Environmental Manual (EM), Radiological Effluent Control Program Manual (RECM), Offsite Dose Calculation Manual (ODCM), and Process Control Program (PCP), remain in RETS.

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: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: April 29, 1996

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 15.3.14, ''Fire Protection System,'' and Section 15.4.15, ''Fire Protection System.'' These specifications would be relocated to other licensee-controlled documents in accordance with Nuclear Regulatory Commission generic guidance. Additional administrative changes consistent with the relocation are also proposed.

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

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

This change request proposes to remove certain fire protection program requirements from the Point Beach Technical Specifications and incorporate them into the Final Safety Analysis Report (FSAR) and the Fire Protection Evaluation Report (FPER). No requirements are eliminated, modified, or de-emphasized by this change. The proposed amendment ensures that any future changes to the fire protection program will be subject to an appropriate evaluation in accordance with NRC regulations to ensure that there are no unreviewed safety questions.

Therefore, these proposed changes are administrative in nature. There are no proposed changes to the physical plant or the processes which ensure the plant's capability to mitigate fires and achieve safe shutdown. Therefore, there is no potential effect on the probability or consequences of previously evaluated accidents.

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

New or different accidents can only be created by new or different accident initiators or sequences. Because there are no proposed changes to the physical plant or the processes which ensure the plant's fire protection capability, new or different kinds of accident initiators will not be introduced by this change. The proposed changes are administrative in nature.

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

The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection. Because there are no proposed changes to the physical plant or the processes which ensure the plant's fire protection capability, there will be no effect on the reactor, reactor containment, or the safety systems which provide their protection. Therefore, the proposed changes will not create a reduction in a margin of safety. The proposed changes are administrative in nature.

Additionally, the proposed revision to Point Beach's operating license will not allow Wisconsin Electric to make changes to the approved fire protection program without prior approval of the Nuclear Regulatory Commission should these proposed changes adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. In accordance with NRC Generic Letter 86-10, any proposed change to the approved fire protection program requires the performance of a 10 CFR 50.59 evaluation and a fire hazards analysis. Should these evaluations indicate that the ability to reach and maintain safe shutdown has been adversely affected, prior NRC review and approval will be obtained prior to effecting the changes. Thus, a significant reduction in a margin of safety cannot occur.

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

Local Public Document Room location: Joseph P. Mann Library, 1516

Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

Date of amendment request: May 16, 1996. This supersedes the October 24, 1995, request published in the Federal Register on November 27, 1995 (60 FR 58409).

Description of amendment request: This license amendment request proposes to revise Surveillance Requirement 4.7.6.e.4 to reflect a proposed design change to the output rating, from 15kW to 5kW, of the charcoal filter adsorber unit heater in the pressurization system portion of the control room emergency ventilation system (CREVS). Surveillance Requirements 4.7.6.c.2, 4.7.6.d, and 4.9.13.b and c, are also being revised to reflect a proposed change to the acceptance criteria for the testing of carbon samples from the CREVS charcoal adsorbers and the auxiliary/fuel building emergency exhaust system charcoal adsorbers. Surveillance Requirement 4.7.7.a for the auxiliary building portion of the auxiliary/fuel building emergency exhaust system is also affected by this proposed change. However, since Surveillance Requirement 4.7.7.a refers to Surveillance Requirements 4.9.13.b and c, no changes to 4.7.7.a are required.

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

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

The design function of the filter adsorber unit heater in the pressurization system portion of CREVS is to reduce the relative humidity of the air entering the charcoal filter beds to 70% relative humidity. Although the original design specified a heater with a rating of 15 kW, review of the design basis calculation for this system indicates that only about 3.13 kW is actually required (including applicable margins to allow for voltage variations). The proposed change to the CREVS heaters— output rating from 15 kW to 5 kW will not affect the method of operation of the system, and the new heater capacity will still exceed filter operational requirements and safety margin. Neither the heater change nor the charcoal testing protocol changes will affect system operation or performance, nor do they affect the probability of any event initiators. These changes do not affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. Therefore, the proposed changes will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR [Updated Safety Analysis Report].

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

The requested change to the CREVS heaters' output rating and the changes to the charcoal sample testing protocol will not affect the method of operation of the systems, and the new heater capacity will still exceed filter operational requirements and safety margin by a significant amount. The proposed changes only affect the heater size in the system and the testing criteria for the charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

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

The requested change to the CREVS heaters' output rating will reduce the heater output of the system, but the new heater output will still exceed filter operational requirements and safety margin by a significant amount. In addition, the reduction in heat load output from the heater will increase the design margin between the cooling capacity of the system air conditioning units and the building heat load. The new charcoal adsorber sample laboratory testing protocol is more stringent than the current testing practice and more accurately demonstrates the required performance of the adsorbers following a design basis LOCA [loss-of-coolant accident]. Therefore, these changes will not reduce the margin of safety of the HVAC [heating, ventilation, and air conditioning] systems' operation.

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

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

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

NRC Project Director: William H. Bateman

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: January 12, 1996, as supplemented March 4, April 3 and April 10, 1996.

Brief description of amendments: The amendments revise the Technical Specification so that the containment integrated leak rate Type A testing will now be performed consistent with the revised 10 CFR Part 50, Appendix J, Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' No changes to implement Option B for the Type B and Type C tests were requested by the licensee at this time.

Date of issuance: May 13, 1996

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

Amendment Nos.: 144 and 138

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 21, 1996 (61 FR 3498); and April 10, 1996 (61 FR 15988) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendments: March 5, 1996

Brief description of amendments: These amendments delete the requirement to perform a pressurizer heater surveillance test and change the requirement for containment visual inspection to prevent sump clogging. These changes are in accordance with selected line items from NRC Generic Letter 93-05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.''

Date of issuance: May 13, 1996

Effective date: May 13, 1996

Amendment Nos. 184 and 178Facility Operating Licenses Nos. DPR-31 and DPR-41: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: April 10, 1996 (61 FR15989) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 13, 1996.No significant hazards consideration comments received: No

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

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

Date of amendment request: January 22, 1996, as supplemented by letter dated April 18, 1996.

Brief description of amendment: The amendment modified the steam generator tube plugging criteria in Technical Specification 3/4.4.5, Steam Generators, and the associated Bases, to allow the implementation of alternate steam generator tube plugging criteria for the tube-to-tubesheet joints (known in the industry as F*) for Unit 1.

Date of issuance: May 14, 1996Effective date: May 14, 1996

Amendment No.: 82

Facility Operating License No. NPF-76: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7553) The additional information contained in the supplemental letter dated April 18, 1996, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 14, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 9, 1996, as supplementedMarch 15, 1996, and April 22, 1996.

Brief description of amendment: The amendment revised the Administrative Controls Section 5.6.6 of the Ginna Technical Specifications to incorporate a reference to the methodology for determining pressure/temperature and low-temperature overpressure protection limits.

Date of issuance: May 23, 1996

Effective date: May 23, 1996

Amendment No.: 64

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7557) The March 15, 1996, and April 22, 1996, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 23, 1996.No significant hazards consideration comments received: No

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

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

Date of application for amendment: February 9, 1996

Brief description of amendment: This amendment changes the setpoints for the steam generator water level-high feedwater isolation function.Date of issuance: May 20, 1996

Effective date: May 20, 1996

Amendment No.: 63

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

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7558) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 20, 1996.No significant hazards consideration comments received: No

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

Saxton Nuclear Experimental Corporation (SNEC), Docket No. 50-146, Saxton Nuclear Reactor Facility (SNEF)

Date of application for amendment: November 21, 1995, as supplemented on March 13, 1996.

Brief description of amendment: The amendment adds GPU Nuclear Corporation as a licensee for the SNEF along with SNEC and transfers all management-related responsibilities for the SNEF from SNEC to GPU Nuclear Corporation.

Date of issuance: May 10, 1996

Effective date: May 10, 1996

Amendment No.: 13Amended Facility License No. DPR-4: Amendment changed the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3502). The Commission also published a notice of consideration of transfer of control of license pursuant to 10 CFR 50.80 on March 19, 1996 (61 FR 11231). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 1996.o significant hazards consideration comments received: No

Local Public Document Room Location: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678

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

Date of application for amendment: December 8, 1995

Brief description of amendment: The amendment revises the Technical Specifications (TS) to: 1) add a new surveillance requirement to 4.1.2.2, 2) delete 3.1.2.3 and 3.1.2.4, revise 3.4.9.3 to assure that only one charging pump is capable of injecting water into the primary coolant whenthe reactor is in a shutdown mode, 4) add a new surveillance requirement to 4.4.9.3, 5) revise the Emergency Core Cooling Water System pump testing acceptance criteria, and 6) revise the BASES supporting the above changes.

Date of issuance: May 10, 1996

Effective date: 30 days after issuance

Amendment No.: 134

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1635) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 10, 1996.No significant hazards consideration comments received: No

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

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: December 19, 1995, as supplemented by letters dated January 5, 1996 and May 3, 1996.

Brief description of amendments: The amendments replace the requirements associated with the control room emergency ventilation system contained in Technical Specification Section 3/4.7.7 with requirements related to the operation of the control room emergency filtration/pressurization system and the control room air conditioning system. In addition, a one-time extension to the allowable outage time for the control room recirculation filtration system is included to facilitate implementation of design modifications.

Date of issuance: May 21, 1996

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

Amendment Nos.: 119 and 111

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

Date of initial notice in Federal Register: January 22, 1996 (61 FR 1637) The January 5, 1996 and May 3, 1996 letters provided clarifying information that did not change the scope of the December 19, 1995, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 21, 1996.No significant hazards consideration comments received: No

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

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: April 23, 1996

Brief description of amendment: The amendment would allow steam generator tubes to remain in service with bands of axial degradation in the tube sheet region, for the remainder of Cycle 11, provided sufficient undegraded tubing remains to satisfy the L*-type criteria restrictions established by the licensee.

Date of issuance: May 20, 1996

Effective date: May 20, 1996

Amendment No.: 110

Facility Operating License No. NPF-8. The amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 19092). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by May 30, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 20, 1996.

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

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 July 5, 1996, 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. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-001, 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).

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

Date of application for amendments: May 15, 1996

Brief description of amendments: The amendment revised Surveillance Requirement (SR) 4.5.2.d.2 in Technical Specification 3/4 5.2 to state that the trisodium phosphate (TSP) contained in the storage baskets in containment is in the form of anhydrous TSP, rather than dodecahydrate TSP, as currently specified.

Date of issuance: May 15, 1996

Effective date: May 15, 1996

Amendment Nos.: Unit 1 - 107; Unit 2 - 99; Unit 3 - 79

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated May 15, 1996.

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

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

NRC Project Director: William H. Bateman

Dated at Rockville, Maryland, this 29th day of May 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

Director, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation

[Doc. 96–13878 Filed 6–4–96; 8:45 am]

BILLING CODE 7590–01–9

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

[Docket No. 301–87]

Notice of Agreement; Monitoring and Enforcement Pursuant to Sections 301 and 306: Canadian Exports of Softwood Lumber

AGENCY: Office of the United States Trade Representative.

ACTION: Notice of monitoring and determination.

SUMMARY: On May 29, 1996, the United States and Canada entered into an agreement on trade in softwood lumber, with effect form April 1, 1996. This agreement is intended to provide a satisfactory resolution to certain acts, policies and practices of the Government of Canada affecting exports to the United States of softwood lumber that were the subject of an investigation initiated by the United States Trade Representative (''USTR'') under section 302(b)(1)(A) of the Trade Act of 1974 (the Trade Act) and that were found to be unreasonable and to burden or restrict U.S. commerce pursuant to section 304(a) on October 4, 1991. The USTR has determined that this agreement will be subject to the provisions of section 306 of the Trade Act and that USTR will monitor Canadian compliance with this agreement pursuant to section 306 of the Trade Act and will take action under section 301(a) if Canada fails to comply with it.

DATES: The U.S.-Canada agreement on trade in softwood lumber was signed on May 29, 1996.

ADDRESSES: Office of the United States Trade Representative, 600 17th Street, NW, Washington, D.C. 20508.

FOR FURTHER INFORMATION CONTACT: Gordana Earp, Deputy Assistant United States Trade Representative for Industry, (202) 395–6160; or William Kane, Associate General Counsel, (202) 395–6800 (for legal issues).

SUPPLEMENTARY INFORMATION: On October 4, 1991, Canada unilaterally terminated a Memorandum of Understanding (MOU) dated December 30, 1986, between the United States and Canada under which, among other things, Canada had imposed a 15 percent export charge on certain softwood lumber products exported to the United States. The MOU had been entered into to settle a pending countervailing duty (CVD) proceeding examining subsidies and injury with respect to imports of Canadian softwood lumber. As of October 4, 1991, Canada ceased collecting export charges under that MOU to offset possible injurious subsidies. In response, on October 4, 1991, (a) the U.S. Department of Commerce announced that it would self-initiate a CVD investigation on softwood lumber from Canada, and (b) the USTR initiated an investigation pursuant to section 302(b)(1)(A) of the Trade Act (19 U.S.C. 2412(b)(1)(A)) and pursuant to section 304(a) of the Trade Act determined that Canada's acts, policies and practices regarding the exportation of softwood lumber to the United States were unreasonable and burdened or restricted U.S. commerce. 56 FR 50738 (October 8, 1991) as amended by 46 FR 58944 (November 22, 1991).

USTR further determined that action was appropriate under section 301 of the Trade Act to restore and maintain the *status quo ante* pending issuance of a preliminary CVD determination, and, if warranted, to impose duties to offset any subsidies found in the investigation. Commerce issued its preliminary CVD determination on March 12, 1992 and its final affirmative CVD determination on May 28, 1992.

Both the domestic industry and affected Canadian parties appealed Commerce's final subsidy determination to binational panels established pursuant to Chapter 19 of the U.S.-Canada Free Trade Agreement (FTA). Following completion of the panel proceedings, and a decision by an Extraordinary Challenge Committee (ECC) established pursuant to FTA Article 1904.13 affirming the results of those proceedings, Commerce—although it expressed disagreement with the panel's findings—on August 16, 1994, revoked the CVD order on softwood lumber from Canada. 59 FR 42029 (Aug. 16, 1994). USTR subsequently terminated the action taken under section 301. 59 FR 52846 (October 19, 1994).

In response to the decisions of the binational panel and the ECC, the domestic industry filed a complaint with the United States Court of Appeals for the District of Columbia Circuit on September 14, 1994, challenging Chapter 19 of the FTA. On December 15, 1994, in order to create a process that could ultimately settle the dispute arising from the unilateral termination in 1991 of the MOU by Canada, and in conjunction with the domestic industry's withdrawal of its challenge to Chapter 19 of the FTA, the United States and Canada agreed to establish a consultative process regarding trade in softwood lumber. The process included the participation of the U.S. Government, Canadian federal and provincial governments, and where appropriate, industries and other interested parties in both countries.

As a result, on May 29, 1996, the United States and Canada entered into an agreement on trade in softwood lumber, with effect from April 1, 1996. During its five-year term, the agreement will foster stable growth in the North American softwood lumber market and ensure fair and competitive trade for U.S. firms and workers by addressing the disruptive effects of unprecedented high levels of Canadian imports previously found by the U.S. Department of Commerce to be subsidized. The agreement requires