Docket No.: STN 50-483

SEP 17 1985

Mr. D. F. Schnell
Vice President - Nuclear
Union Electric Company
P. O. Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

Subject: FEDERAL REGISTER NRC Bi-Weekly Notices of Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations - Callaway Plant, Unit 1

Enclosed is a copy of the Federal Register NRC Bi-Weekly Notices of Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated September 11, 1985.

A proposed amendment notice concerning the revision of Technical Specification 4.6.1.2 and its associated bases regarding containment leakage surveillance requirements to proved clarifications on the leak rate testing of valves pressurized with fluid from a seal system is contained on Page 37091 of this publication.

Sincerely,

(S)

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
Federal Register
dated September 11, 1985

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

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NUCLEAR REGULATORY COMMISSION

Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law (Pub. L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing this regular bi-weekly notice. Pub. L. 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 bi-weekly notice includes all amendments issued, or proposed to be issued, since the date of publication of the last bi-weekly notice which was published on August 28, 1985 (50 FR 34933), through August 30, 1985.

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR 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 amendments 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. The Commission will not normally make a final determination unless it receives a request for a hearing.

Comments should be addressed to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

By October 11, 1985, 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 petition for leave to intervene. Requests for a hearing and petitions 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. 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 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 fifteen (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 fifteen (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, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. 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 involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

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 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.

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, D.C. 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., by the above date. Where petitions are filed during the last ten (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 (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to *(Branch Chief)*: 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 Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding 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, 1717 H Street, NW., Washington, D.C., and at the local public document room for the particular facility involved.

Arizona Public Service Company et al., Docket No. STN 50-528, Palo Verde Nuclear Generating Station, Unit No. 1, Maricopa County, Arizona; Date of amendment request: August 5, 1985.

Description of amendment request: The amendment would permit a one-time exception for approximately 24 hours to Technical Specifications 3.4.1.2, 3.4.1.3, and 3.7.1.6, involving the reactor coolant system pumps and the atmospheric dump valves, to allow the performance of the Natural Circulation Cooldown Test.

Basis for proposed no significant hazards consideration determination: The Commission has provided certain examples (48 FR 14870) of actions likely to involve no significant hazards considerations. One of the examples (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. The proposed amendment involved here is similar in that the exception request may reduce a safety margin in some way for a limited time only during the performance of the Natural Circulation Cooldown Test, but the results of an analysis of the planned test are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. Accordingly, the Commission proposes to determine that this change does not involve significant hazards considerations.

Local Public Document Room location: Phoenix Public Library, Business, Science and Technology Department, 12 East McDowell Road, Phoenix, Arizona 85004.

Attorney for licensees: Mr. Arthur C. Gehr, Snell & Wilmer, 3100 Valley Center, Phoenix, Arizona 85007.

NRC Branch Chief: George W. Knighton.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland; Date of application for amendments. June 28, 1985

Description of amendment request: The proposed amendments would change the Unit 1 and Unit 2 Technical Specifications (TS) to: (1) Reflect a clarification of requirements associated with the containment purge isolation valves in TS Table 3.3-3, "Engineered Safety Features Actuation System Instrumentation," and TS Table 3.6-1, "Containment Isolation Valves," (2) modify TS 3.9.4, "Containment Penetration," to allow the use of an alternate closure for the emergency personnel escape lock, (3) delete TS 6.13, "Environmental Qualifications," and (4) correct identified spelling errors and changes in terminology.

Basis for no significant hazards consideration determination: BG&E has requested a change to TS Tables 3.3-3 and 3.6-1 to correct an inconsistency between operability requirements for the containment purge isolation valves and related requirements.

The containment purge isolation valves allow outside air to enter the containment and vent the containment atmosphere to the environment. At the present time, these valves are required to be isolated by the requirements of TS 3.6.1.7, "Containment Purge System," to prevent these valves from being opened, during Modes 1 through 4 (power operation through hot shutdown). Furthermore, the purge isolation valves are required to be operable, meaning capable of automatic closure to a leak-tight condition, by the requirements of TS 3.9.9, "Containment Purge Valve Isolation System," during core alterations or movement of irradiated fuel inside the containment (during Mode 6). A comparison of the requirements of TS 3.6.1.7 and TS 3.9.9 indicates that the containment purge isolation valves may be inoperable (open) in Mode 5 (cold shutdown) or in Mode 6 (refueling) when neither core alternations nor movement of irradiated fuel inside containment is underway or any time in Mode 6 when the containment purge valves are closed. Conditions under which the containment isolation valves may be inoperable are consistent with conditions under which containment (leak-tight) integrity is not required per TS 3.6.1.1 and TS 3.9.4.

BG&E has identified two inconsistencies regarding requirements associated with the containment purge isolation valves. The first instance of inconsistency involves TS Table 3.6–1. In this case, the TS requires that the valve isolation response time for the containment purge isolation valves be applicable ". . . for Mode 5 and 6 during which time these valves may be opened." Since isolation response times should not be applicable when the valves are not required to be operable, BG&E has proposed to reword the applicability of the response time to be ". . . in Mode 6 when the valves are required operable and they are open." This proposed applicability wording is consistent with operability requirements of the containment purge isolation valves.

The second instance of inconsistency involves TS Table 3.3–3 which specifies operability requirements for devices for manual and automatic closure of the containment purge isolation valves. At the present time, the purge valve control switches for manual closure must be operable in Modes 5 and 6 and the containment radiation-high area monitor (for automatic closure) must be operable in Mode 6. The licensee has proposed changing the operability requirements for these closure devices to ". . . in Mode 6 when the valves are required operable and they are open."

The proposed changes to TS Tables 3.6–1 and 3.3–3 would not increase the probability or consequences of any accidents for which closure of the containment purge insolation valves is required. Operability of automatic and manual valve closure devices and closure response times would be applicable at all times when the containment purge isolation valves are required to be operable (these requirements only apply to the containment purge isolation valves). No new or different type of accident would be created by the proposed TS changes since no new operational modes are being created for the containment purge isolation valves. Finally, no safety margins are reduced since no operational or design changes are proposed. Accordingly, the Commission proposes to determine that the proposed changes to TS Tables 3.3–3 and 3.6–1 involve no significant hazards considerations.

The licensee has proposed a change to TS 3.9.4b which would provide a footnote to allow use of a temporary closure for the containment emergency personnel escape lock during refueling activities. At the present time, TS 3.9.4 requires at least one door in each air lock to be closed during core alterations or movement of irradiated fuel inside the containment.

The personnel escape lock described in Section 5.1.2.1 of the Calvert Cliffs Final Safety Analysis Report (FSAR) is located at elevation 49'4" and is provided with outer and inner doors. During refueling operations, the licensee proposes to replace the inner personnel escape lock door with a temporary closure; the outer door would remain open at this time. This temporary closure would contain several penetrations to facilitate work inside containment, during core alterations or movement of irradiated fuel, when containment integrity is required. The licensee has indicated that the temporary closure and its penetrations meet the applicable design requirements of the permanent door for use during reactor Mode 6 (refueling). Installation and leak testing of the temporary closure would be controlled by a plant procedure.

The Bases for TS 3/4.9.4 states the following with regard to containment closures such as the personnel escape lock: "The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE." Since no containment pressurization results from the design basis (fuel handling) event during Mode 6, containment closures need only be vapor-tight rather than capable of withstanding excess pressure. Since the temporary closure is fabricated to standards equivalent to the personnel escape lock for Mode 6 utilization and installation and testing will be in accordance with plant procedures, the temporary closure can be expected to perform in a manner equivalent to that of the personnel escape lock door during the design basis event in Mode 6. Accordingly, the proposed TS does not involve any increase in the probability or consequences of accidents previously considered. Moreover, since either a personnel escape lock door or the equivalent (temporary closure) will be in place during core alterations or movement of irradiated fuel inside containment, no new or different type of accident will be created. Finally, since the temporary closure performs in a manner equivalent to the permanent personnel escape lock door during Mode 6, no safety margin with regard to off-site dose following the design basis Mode 6 event will occur. Based upon these conclusions, the Commission proposes to determine that the proposed change to TS 3.9.4b, to allow use of a temporary closure device, involves no significant hazards considerations.

The licensee has proposed to delete TS 6.13, "Environmental Qualifications." This TS provides schedule requirements for completion of activities relating to environmental qualification of electrical equipment important to safety that have already passed. Moreover, environmental qualification requirements, including schedules, are incorporated in 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear power plants," and thus need not appear in the TS.

On April 6, 1983, the NRC published guidance in the **Federal Register** (48 FR 14870) concerning examples of amendments that are not likely to involve a significant hazards consideration. One such example, (i), provides for "A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature." Deletion of the superseded requirements of TS 6.13 is within the scope of example (i).

Accordingly, the Commission proposes to determine that the proposed deletion of TS 6.13 involves no significant hazards considerations.

Finally the licensee proposes to correct 14 spelling and one terminology error in the TS as detailed in the June 28, 1985 application. As indicated by example (i), these changes to the TS are not likely to involve a significant hazards consideration. Accordingly, the Commission proposes to determine that the correction of the spelling and terminology errors identified in the June 28, 1985 application involves no significant hazards considerations.

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

Attorney for licensee: George F. Trowbridge, Esq., Shaw, Pittman, Posts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Unit Nos. 1 and 2, Benton County, Illinois

Date of application for amendments: June 25, 1985.

Description of amendments request: These amendments would delete items A.1 and A.2 of the Commission's February 29, 1980 Confirmatory Order for the Zion Nuclear facility. Item A.1 of that Order was a restriction on the power level to maintain a calculated peak clad temperature of 2050°F under the conditions of the 10 CFR Part 50, Appendix K analysis submitted by licensee on October 22, 1979; Item A.2

was a restriction on load following maneuvers.

Basis for proposed no significant hazards consideration determination: 10 CFR 50.92 states that a proposed amendment will involve a no significant hazards consideration if the proposed amendment does 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.

In accordance with 10 CFR 50.91(a)(1), the licensee submitted the following analysis of the amendment request using the standards in 10 CFR 50.92.

Criterion 1—A Significant Increase in the Probability of an Accident

Proposed Deletion of Item A.1

The consequences of a LOCA are not significantly increased by the deletion of the 2050°F limit and the subsequent adherence to the 220°F limit of 10 CFR 50.46 since this type of change to peak clad temperature during a large break LOCA would have no relationship to the probability of the initiating pipe break.

Proposed Deletion of Item A.2

Attachment 3 to the June 25, 1985 submittal demonstrated that there was no observable correlation between load changes above 50% and the frequency of reactor trips. Thus, the probability of an accident is not affected.

The FSAR already assumes conservative core parameters for the accident analyses. These values (T_{avg}, Pressure, etc.) will still bound actual core conditions during load changes. Thus, the consequences of any postulated accident are unchanged. While load follow maneuvers may have been considered to represent an additional risk in 1980, the experience gained since that time has shown that these load changes are routine in nature.

Criterion 2—A New or Different Kind of Accident

Proposed Deletion of Item A.1

The parameter referred to as the peak cladding temperature is the maximum calculated temperature which could be reached during the course of an already ongoing LOCA. Therefore, the increase in this parameter in and of itself is not an initiator of another unrelated accident. Thus, the 150°F increase in allowed peak cladding temperature during a LOCA does not create the possibility of a new or additional accident.

Proposed Deletion of Item A.2

The performance of load follow maneuvers affects a number of reactor parameters. Exemples of these parameters include rod position, boron concentration, and power distribution. Limiting combinations of these and other parameters have been considered in Zion initial design and each cycle-specific reload safety analysis.

In addition, all equipment necessary to accomplish changes in Zion power level has been originally designed for load following functions. Therefore, these power changes do not create the possibility of a new or different kind of accident not previously considered.

Criterion 3—A Significant Reduction in Margin of Safety

Proposed Deletion of Item A.1

The 150 °F increase in allowable peak clad temperature does not involve a significant reduction of safety margins. Attachment 2 to the application demonstrates the calculational superiority of the current anlysis over the methodology used for the October 22, 1979 submittal. It also demonstrates that Zion will be in compliance with 10 CFR 50.46 and the Standard Review Plan.

Proposed Deletion of Item A.2

Zion's safety analysis utilizes conservative core parameters as initial accident conditions. These values will still bound actual conditions during load changes. Thus, there has been no reduction in safety margins.

Therefore, since the application for amendment satisfies the criteria specified in 10 CFR 50.92, Commonwealth Edison Company has made a determination that the application involves no significant hazards consideration.

The staff has reviewed the licensee's analysis and agrees with the conclusion that the proposed amendment satisfies the three standards of 10 CFR 50.92 for no significant hazards consideration.

In addition, the Commission has provided guidance concerning the application of the standard for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). The examples of actions involving no significant hazards that apply to each of the proposed changes are discussed below.

Proposed Deletion of Item A.1

A new LOCA ECCS analysis for Zion was submitted by Commonwealth Edison Company in October 1984, then supplemented in February and April 1985. This new analysis, which was approved by the Commission on May 24, 1985, incorporates a number of methodological and calculational improvements over the analysis submitted on October 22, 1979. The new analysis reflects improved code interfacing for core reflooding rate data. It uses a BART computer model, a mechanistic core heat transfer model that represents an improvement over heat-up methods used in the 1979 analysis. It also accounts for increases in steam generator tube plugging, as well as assumes a higher total peaking factor for core power distribution prior to the postulated design basis LOCA.

In general, the new analysis, in addition to being calculationally superior to the methodology used for the October 22, 1979 submittal, also demonstrates that Zion will be in compliance with 10 CFR 50.46 and the Standard Review Plan.

Thus, example (vi) of the Commission's guidance for determining there are no significant hazards is applicable in this instance. Example (vi) reads as follows:

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: For example, a change resulting from the application of a small refinement of a Previously used calculational model or design method.

Since what is involved here is a refinement to the calculational method and the results using this method are still in compliance with the Standard Review Plan, staff proposes to make a no significant hazards consideration determination for Item A.1.

Proposed Deletion of Item A.2

The October 1984 submittal by licensee discusses the operating stability of the Zion units while performing load changes above 50% power. As discussed therein, the excellent stability of the load follow mode of operation is demonstrated by the historical information presented. While load follow maneuvers may have been considered to represent an additional risk in 1980, the experience gained since that time has shown that these load changes are routine in nature.

Thus example (iv) of the Commission,s guidance regarding significant hazards derterminations is applicable in this instance. Example (iv) reads as follows:

(iv) A relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a satisfactory way that the criteria have been met.

Based on this criteria, staff proposes to make a no significant hazards consideration determination for Item A.2 since a lesser degree of risk has been demonstrated since the original review.

Local Public Document Room location: Zion-Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

Attorney to licensee: P. Steptoe, Esq., Isham, Lincoln and Beale, Counselors at law, Three First National Plaza, 51st Floor, Chicago, Illinois 60602.

NRC Branch Chief: Steven A. Varga.

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Unit Nos. 1 and 2, Benton County, Illinois

Date of application for amendments: July 12, 1985, supplemented July 26, August 2 and 7, 1985.

Description of amendments request: These amendments would raise the enrichment limits to approximately 3.7 w/o for the spent fuel pool and 4.0 w/o for the new fuel vault.

Basis for proposed no significant hazards consideration determination: 10 CFR 50.92 states that a proposed amendment will involve a no significant hazards consideration if the proposed amendment does 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.

In accordance with 10 CFR 50.91(a)(1) the licensee submitted the following analysis of the amendment using standards in 10 CFR 50.92.

Criterion 1

The storage of 3.7 w/o fuel in the Spent Fuel Pool and 4.0 w/o fuel in the New Fuel Vault could only affect the fuel handling accidents. The enrichment increase will not significantly affect the potential consequences of a fuel drop accident, since the isotopic content of a discharged assembly is relatively insensitive to the assembly's initial enrichment.

The probability of a fuel handling accident is similarly unaffected by the enrichment increase. There are no structural changes involved that could affect the handling characteristics of Zion's fuel.

Criterion 2

The enrichment increase does not create the possibility for any new or different type of accident. All other acceptance criteria and operating parameters (DNBR, F_Q, etc.) will remain unchanged

Criterion 3

While the storage of fuel with increased enrichment will potentially bring the Spent Fuel Pool and the New Fuel Vault somewhat closer to criticality than was previously possible, this safety margin reduction is not significant.

Attachments 2 and 4 to the July 12, 1985 submittal demonstrate that the results of the proposed change are clearly within all acceptable criteria. Specifically, the reactivity acceptance criteria of the Standard Review Plan, Section 9.1.1 and 9.1.2, have been satisfied.

Therefore, since the application for amendment satisfies the criteria specified in 10 CFR 50.92, Commonwealth Edision Company has made a determination that the application involves no significant hazards consideration.

The staff has reviewed licensee's analysis and concludes that the amendment satisfies the three criteria listed in 10 CFR 50.92. Based on that conclusion the staff proposes to make no significant hazards consideration determination.

Local Public Document Room location: Zion-Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

Attorney to licensee: P. Steptoe, Esq., Isham, Lincoln and Beale, Counselors at Law, Three First National Plaza, 51st Floor, Chicago, Illinois 60602.

NRC Branch Chief: Steven A. Varga.

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Unit Nos. 1 and 2, Benton County, Illinois

Date of application for amendments: August 8, 1985.

Description of amendments request: These amendments would modify Technical Specifications to reflect installation of a degraded grid voltage protection system.

Basis for proposed no significant hazards consideration determinction: 10 CFR 50.92 states that a proposed amendment will involve a no significant hazards consideration if the proposed amendment does 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.

In accordance with 10 CFR 50.91(a)(1) the licensee submitted the enclosed analysis of the proposed amendment using the Standard in 10 CFR 50.92.

Criterion 1

The installation of degraded grid voltage protection provides additional assurance that a stable source of power for the required safety related equipment will be available. This increases the probability that the equipment will be capable of performing the required function. Thus, the probability and consequences of the previously analyzed accidents have not been increased.

Criterion 2

Reference (a) to the August 8, 1985 application established the design criteria that spurious operation of the degraded grid protection system would not occur. Reference (b) to the August 8, 1985 application stated that this design goal has been met. Thus, this modification can only serve to enhance the power supply's reliability and does not create the possibility of a new type of accident.

Criterion 3

The margin of safety is increased by this change. As discussed above, the safety-related power supply should be more reliable when protected against a degraded grid voltage.

Therefore, since the application for amendment satisfies the criteria specified in 10 CFR 50.92 Commonwealth Edison Company has made a determination that the application involves no significant hazards consideration

The staff has reviewed the licensee's analysis and concludes that the proposed amendment satisfies the criteria of 10 CFR 10.92. Based on that conclusion staff proposes to make a no significant hazards consideration determination.

Local Public Document Room location: Zion-Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

Attorney to licensee: P. Steptoe, Esq., Isham, Lincoln and Beale, Counselors at Law, Three First National Plaza, 51st Floor, Chicago, Illinois 60602.

NRC Branch Chief: Steven A. Varga.

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

Date of amendment request: August 2, 1985.

Description of amendment request: The proposed amendment would revise the Indian Point Nuclear Generating Unit No. 2 Technical Specifications to delete the Boron Injection Tank (BIT) and its associated limiting conditions for operation and surveillance requirements. Consolidated Edison has requested the elimination of the BIT in order to remove it as a source of operational and maintenance problems at Indian Point 2.

Basis for proposed no significant hazards consideration determinations: Consistent with the Commission's criteria for determining whether a proposed amendment to an operating license involves no significant hazards considerations, 10 CFR 50.92 (48 FR 14871), the proposed revisions to the Technical Specifications will not involve a significant increase in the probability or consequences of an accident previously evaluated because the licensee's analyses show that for all Final Safety Analysis Report (FSAR) cases, the Departure from Nucleate Boiling Ratio (DNBR) is above the limiting value of 1.30. The removal of the BIT will not create the possibility of a new or different kind of accident previously evaluated becasue the function of the BIT is to provide a source of concentrated boric acid to provide additional shutdown margin following a main steam line break. No new or different accidents would be involved. The removal of the BIT will not involve a significant reduction in margin of safety because, as stated above, the DNBR will remain above the minimum value of 1.30.

Therefore, the staff proposes to determine that the amendment does not involve a significant hazards determination.

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Branch Chief: Steven A. Varga.

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

Date of amendment request: June 27, 1985.

Description of amendment request: The amendment proposes changes to several Technical Specification surveillance frequencies. These surveillances are currently required to be performed at various intervals regardless of scheduled refueling shutdowns. These changes will allow continued operation of the plant between refueling shutdowns without having to take the plant to Cold Shutdown until a scheduled refueling outage. Standard Technical Specifications (STS) surveillance frequencies for equivalent equipment were used to establish these new surveillance frequencies.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards of 10 CFR 50.92 by providing certain examples (48 FR 14870, April 6, 1983). One of the examples (vi) of actions not likely to involve significant hazards considerations relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP).

The proposed changes to the surveillance requirements for the reactor safety system scram circuits, containment sphere isolation trip circuits, and emergency condenser trip circuits maintain the existing surveillance frequency of, each refueling outage, but change the bounding limit from 12 months to 18 months. This type of surveillance frequency currently exists in several other sections of the Big Rock Point Technical Specifications. The surveillance and replacement frequencies for the squib primers and trigger assemblies (liquid poison system components) are proposed to be changed from, at least every 12 months and replaced every 24 months, to, at least every 18 months and replaced every 36 months. Therefore, the surveillance testing and replacement frequencies are proposed to be decreased. The licensee maintains, however, that although these frequencies are being decreased, they do continue to limit the longest service of the components to the manufacturer's limit of 5 years.

Additionally, a smiliar change is proposed to the surveillance frequency for functional testing of the control rod permissive circuits. The current surveillance frequency provided in Section 6.2.2 and in Section 7.6 of the existing Technical Specifications requires functional testing of the permissive circuits to be "not less frequent than once every 12 months". The proposed change allows the functional testing to be performed no less frequent than every 18 months. Also, since the capability exists to accomplish this testing while at power, there is no need to tie the surveillance frequency to a refueling shutdown (as does current technical specifications). Consequently, the proposed change provides for testing prior to each major refueling shutdown.

Although the surveillance frequencies of all of these tests do increase, the proposed changes are consistent with existing surveillance frequencies established in BWR STS. These proposed changes, therefore, fit example (vi), described above, since the changes are clearly within all acceptable criteria with respect to the system or component specified in the SRP. The SRP specifies that BWR STS are consistent with the regulatory guidance contained in the SRP. On this basis, therefore, the staff proposes to determine that the requested changes do not involve a significant hazards consideration.

Local Public Document Room location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

NRC Branch Chief: John A. Zwolinski.

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

Date of amendment request: July 19, 1985; supersedes March 10, 1982 application.

Description of amendment request: The submittal received proposes changes to Big Rock Point (the facility) Technical Specifications (TS) for the Plant Monitoring System. The submittal also includes proposed TS for the addition of the Containment Pressure and Containment Water Level Monitors. This submittal completes Consumers Power Company's (the licensee's) commitment and associated responses to these specific NUREG-0737 items. The original application was initially noticed in the **Federal Register** on August 23, 1983 (48 FR 38397).

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870, April 6, 1983). One of the examples (ii) of actions not likely to involve significant hazards considerations relates to a change that constitutes an additional limitation, restriction, or control not presently included in the TS. Section 6.4 of the current TS provides the list of systems which are considered part of the plant monitoring system. This proposed change adds to that list the containment pressure and water level monitoring systems. The proposed new Section 6.4.4 and changed Section 7.6 provide the new limiting conditions for operation, associated actions, and surveillance requirements for these new systems. The addition of these systems and associated operability requirements to the facility TS provide additional limitations and restrictions and therefore fit example (ii) described above. On this basis, the staff proposes to determine that the requested TS changes would involve no significant hazards considerations.

Local Public Document Room location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

NRC Branch Chief: John A. Zwolinski.

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

Date of amendment request: August 16, 1985.

Description of amendment request: The application requests changes be made to the Technical Specifications (TS) in support of the planned fuel reloading. Specifically, the reload I1 fuel Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits are to be changed. The changes are required in order to implement new reactor operating limits for reload I1 fuel and facility power operation.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (April 6, 1983, 48 FR 14870). One of the examples (iii) of actions not likely to involve significant hazards considerations relates to a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the TS, that the analytical methods used to demonstrate conformance with the TS and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

The proposed TS changes will implement reactor operating limits for reload I1 fuel. These reactor operating limits are based on the Loss of Coolant Accident (LOCA) analysis required by 10 CFR 50.46. The MAPLHGR limits for reload I1 are based on the LOCA analysis submitted by Consumers Power Company letter dated March 7, 1979 (Exxon Nuclear Company (ENC) Report XN–NF–78–53). Reload I1 is identical to the previous G3/G4 reloads in all respects except as described in the subsequent paragraph. The G3/G4 type reload TS changes were previously evaluated by the staff and, as a result, the staff issued Amendment No. 44 to the Facility Operating License (DPR–6).

Reload I1 fuel is identical in all respects to G3/G4 reloads except the reload I1 fuel has a smaller pellet-to-clad gap. The reload I1 fuel also has a higher helium prepressurization than the previous G3/G4 fuels. These minor differences have no effect on the thermal hydraulic design basis for ENC I1 fuel.

Therefore, since the reload does not contain any fuel assemblies significantly different from those previously found acceptable to the NRC, these proposed TS changes fit example (iii) described above.

Another example (i) of actions not likely to involve a significant hazards consideration relates to a change which is purely administrative, a change to achieve consistency throughout the TS, correction of an error, or a change in nomenclature. The proposed column heading changes to Tables 1 and 2 of the facility TS are editorial in that they provide space for the additional column required for the reload I1 fuel Operating Limit TS described above, and therefore fit example (i).

On these bases, therefore, the staff proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

NRC Branch Chief: John A. Zwolinski.

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: September 6, 1984.

Description of amendment request: The proposed amendments would add Technical Specification 3/4.4.11 and an associated bases to provide operability and surveillance requirements for the Reactor Coolant System (RCS) Vent System required by 10 CFR 50.44(c)(3)(iii).

Basis for proposed no significant hazards consideration determination: The RCS vent system is a newly installed system required by 10 CFR 50.44(c)(3)(iii) to provide for post-accident venting of noncondensible gases from the high points of the reactor coolant system. The McGuire design provides vent paths from the pressurizer steam space and the reactor vessel head.

The Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards considerations by providing certain examples published in the **Federal Register** on April 6, 1983 (48 FR 14870). One of the examples of an amendment likely to involve no significant hazards consideration relates to changes (ii) that constitute additional limitations, restrictions, or controls not presently included in the Technical Specifications. The proposed amendments of the Technical Specifications match the example because they would impose additional imitations for operation and additional surveillance requirements for a newly installed system not presently addressed in the Technical Specifications. The proposed addition does not replace or relax any existing requirements in the Technical Specification. Therefore, the Commission proposes to determine that the proposed amendments do not involve a 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, P.O. Box 33189, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Branch Chief: Elinor G. Adensam.

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

Date of amendment requests: November 12, 1984, and January 30, 1985.

Description of amendment request: The proposed amendments would change the action statement for the limiting condition for operation, and the surveillance requirement, for Technical Specification 3/4.5.1, Cold Leg Injection Accumulators. Specifically the requirement to be in hot shutdown (specified in the action statement when one accumulator is inoperable for reasons other than a closed isolation valve) would be replaced by a requirement to reduce pressurizer pressure to less than 1000 psig. The requirement to be in hot standby within "1 hour and in HOT SHUTDOWN within the following 12 hours" (specified in the action statement when one accumulator is inoperable due to the isolation valve being closed) would be changed to require that the reactor be in hot standby within "6 hours and that pressurizer pressure be reduced to less than 1000 psig within the following 6 hours." Surveillance requirement 4.5.1.1.1.d, which requires periodic testing of the automatic opening feature of the accumulator isolation valves, would be deleted.

Basis for proposed no significant hazards consideration determination: Technical Specification 3.5.1.1 requires each cold leg injection accumulator to be operable with the isolation valve open when pressurizer pressure is above 1000 psig. The existing Specification 3.5.1.1 allows 1 hour to place the reactor in hot standby when accumulator inoperability is due to a closed isolation valve, but allows 6 hours when accumulator inoperability is not due to a closed isolation valve. This is inconsistent because the potential causes for accumulator inoperability other than a closed accumulator isolation valve (e.g., total loss of nitrogen gas pressure) have a safety significance comparable to that of a closed accumulator isolation valve. The 1 hour requirement is unnecessarily conservative since the inoperability of the accumulators for up to 6 hours was previously determined to pose negligible adverse safety consequence. Accordingly, the change from 1 hour to 6 hours to be in hot standby when inoperability is due to a closed isolation valve is equally acceptable and does not cause a significant adverse effect upon the probability or consequences of an accident previously evaluated, does not give rise to any new accident, and has no significant adverse impact upon a safety margin.

The other change to Specification 3.5.1.1 would require that pressurizer pressure be lowered below 1000 psig within 6 hours instead of placing the reactor in hot shutdown as currently required. Plant operating procedures require that the accumulators be isolated below a reactor coolant system pressure of 1000 psig in order to prevent inadvertent injection during planned depressurization (i.e., shutdown). In support of these operating procedures, licensee's analysis of a large break LOCA during a plant cooldown has previously demonstrated (see Supplement 2 to SER, Section 6.3.4) that adequate protection is provided without the cold leg injection accumulators if reactor coolant system pressure at the time of the accident was at or below 1000 psig. Thus, because accumulators serve no safety funtion below 1000 psig, the change does not adversely affect either the probability or consequences of an accident previously evaluated, does not give rise to any new accident, and has no adverse impact on a safety margin.

The accumulator isolation valves must be open for the accumulators to accomplish their safety (injection) function. The design of the control circuit for the motor-operated accumulator isolation valve as accepted by the staff in SER Section 7.3.3 protected against inadvertent closure of the valve by an automatic opening feature. Although the valve is normally open when RCS pressure is above 1000 psig, it receives a safety injection signal to override any bypass feature and cause automatic opening should the valve be closed. The licensee proposes to delete Surveillance Specification 4.5.1.1.1.d which requires periodic testing of the automatic opening feature of the accumulator isolation valves because changes in operating procedures negate the need for (and functioning of) such a feature (and hence the need for its testing). The licensee's operating procedure for unit startup requires that the valves be opened before exceeding 1000 psig, and that after opening, power to the valve operators is to be disconnected by removal of the breaker from the circuit. Hence, the possibility of inadvertent closure is eliminated by removal of the power source at all times except for those brief periods during planned startups and shutdowns when a deliberate change in valve position is required. The possibility of prolonged operation following inadvertent failure to open the isolation valves during repressurization of the reactor coolant system in accordance with the licensee's startup procedures is eliminated by Surveillance Specification 4.5.1.1.1.a(2) which would not be changed by the proposed amendments and requires verification at least once per 12 hours that each accumulator isolation valve is open. Elimination of the periodic test requirement where the function to be tested is no longer relied upon, and where the 12-hour surveillance requirement is retained, would not have a significant adverse effect on either the probability or consequences of any accident previously evaluated, give rise to any new accident mechanisms, or significantly reduce any safety margin.

The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870) of actions involving no significant hazards considerations. The proposed changes do not match any of the examples. However, based upon our preliminary review of the amendment requests and the above discussion, the Commission proposes to determine that operation of the facility in accordance with the proposed amendments 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. Therefore, the staff proposes to determine that this request involves no significant hazards consideration.

The licensee's letter of January 30, 1985, also requested changes to Technical Specfications for the Upper Head Injection Systems. This part of the request is outside the scope of this notice.

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, P.O. Box 33189, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Branch Chief: Elinor G. Adensam.

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: January 30, 1985.

Description of amendment request: The proposed amendments would revise action statements of the limiting conditions for operation and a surveillance requirement in Technical Specificiation 3/4.5.1.2, Upper Head Injection Accumulator System (UHI). Specifically, the requirement to be in hot shutdown (specified in ACTION (a) which applies when the UHI is inoperable for reasons other than a closed isolation valve) would be replaced by a requirement to reduce presur-izer pressure to less than 1900 psig. The requirement to be in "HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours" (specified in ACTION (b) which applies when the UHI is inoperable due to a closed isolation valve) would be changed to require that the reactor be in "at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1900 psig within the following 6 hours." Surveillance Specification 4.5.1.2.c(1) would be clarified to more accurately reflect the type of testing used to verify auto-matic closure of each UHI accumulator isolation valve (i.e., to reflect use of "an actual or simulated water level signal") and to clarify that "if actual water level is used, then the accumulator should be at atmospheric pressure."

Basis for Proposed No Significant Hazards Consideration Determination: Technical Specification 3.5.1.2 requires each UHI to be operable with the isolation valves open when pressurizer pressure is above 1900 psig (i.e., for Modes 1, 2, and 3, but for Mode 3 only above 1900 psig). The existing Specification 3.5.1.2 allows 1 hour to place the reactor in hot standby when UHI is inoperable due to a closed isolation valve (i.e., for ACTION (b)), but allows 7 hours when UHI inoperability is not due to a closed isolation valve (i.e., for ACTION (a)). This is inconsistent because the potential causes for UHI inoperability other than a closed isolation valve (e.g., total loss of the gas-bearing accumulator pressure) have a safety significance comparable to that of a closed isolation valve. The 1 hour requirement is unnecessarily conservative since the inoperability of UHI for up to 7 hours was previously determined to pose negligible adverse safety consequences. Therefore, the change from 1 to 6 hours to be in hot standby when UHI inoperability is due to a closed isolation valve is also acceptable and will not significantly increase the probability or consequences of accidents previously evaluated, will not create any new accident, and will not have a significant adverse effect upon safety margins.

The other change to ACTION (a) and ACTION (b) would permit the pressurizer pressure to be reduced below 1900 psig in operational Mode 3 (hot standby) instead of placing the reactor in hot shutdown. (for ACTION (b), an additional conservatism would be introduced in that the change would require that this pressure reduction be achieved within 12 hours, whereas the existing ACTION (b) provides a total period of 13 hours for the plant to be in hot shutdown.) Plant operating procedures require that the UHI isolation valves be closed below a reactor coolant system pressure of 1900 psig in order to prevent inadvertent injection during planned depressurization (i.e., shutdown). In support of these operating procedures, licensee's analysis of a large break LOCA during a plant cooldown has previously demonstrated (see Supplement 2 to SER, Section 6.3.4) that adequate protection is provided without UHI injection if reactore coolant system pressure at the time of the accident was at or below 1900 psig. Thus, because UHI serves no safety function below 1900 psig, the change does not adversely affect either the probability or consequences of an accident previously evaluated, does not give rise to any new accident, and has no adverse impact on a safety margin.

The current Surveillance Specification 4.5.1.2.c requires that each UHI accumulator isolation valve be periodically verified to close automatically when the water level is 76.25 ± 3.3 inches above the bottom inside edge of the water filled accumulator with atmospheric pressure in the accumulator. The specification requires clarification because in its present form it could be interpreted to mean that the actual tank water level is to be reduced to the setpoint in order to verify that each accumulator isolation valve closes. Such a limited interpretation is not intended; use of simulated signals to test safety systems in which an instrument reaching a setpoint actuates a devise is an industry-wide practice which is also acceptable to the Commission as evidenced by its acceptance for other safety related systems involved with water level (e.g., high pressurizer water level and low steam generator water level). Surveillance Specification 4.5.1.2.C would be modified to clarify that simulated signals may be used to verify automatic accumulator isolation valve closure.

The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples of actions invloving no significant hazards considerations, (i), relates to purely administrative changes to the Technical Specifications. The change to Surveillance Specification 4.5.1.2.c matches the example because it merely clarifies the testing requirement consistent with the Commission's intended meaning. The changes to Specification 3.5.1.2 do not match any of the examples. However, based upon our preliminary review of the amendment requests as reflected in the above discussion, the Commission proposes to determine that operation of the facility in accordance with the proposed amendments would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Therefore, the staff proposes to determine that this request involves no significant hazards consideration.

The licensee's letter of January 30, 1985, also requested changes to Technical Specifications with respect to the UHI membrane located between the water-filled and nitrogen bearing accumulators. This part of the request is outside the scope of this notice.

Local Public Document Room locations: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for Licensee: Mr. Albert Carr, Duke Power Company, P.O. Box 33189, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Branch Chief: Elinor G. Adensam.

Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: July 12, 1985.

Description of amendment request: The proposed amendment would revise Section 3.2.3, "Nuclear Enthalpy Hot Channel Factor", by deleting the rod bow penalty multiplier. The basis of the requested change is contained in a Westinghouse Topical Report, WCAP-8691, Revision 1, which the staff has approved on December 29, 1982.

The report provides a basis for removing the rod bow penalty applied to the nuclear enthalpy hot channel factor by the use of a rod bow power peaking factor uncertainty. This is then statistically combined with the nuclear power distribution uncertainty and the engineering hot channel factor, to yield a new total heat flux hot channel factor uncertainty with a maximum value of 1.069. This value is the maximum required total peaking factor uncertainty, including the effects of rod bow.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for derterming whether a significant hazards consideration exists by providing a certain examples (48 FR 14870). One of these, Examples (vi), involving no significant hazards consideration is "a change resulting from the application of a small refinement of a previously used calculational model or design method." The requested change matches this example since the rod bow calculational model has been refined by the NRC-accepted methodology provided in WCAP-8691 Revision 1. The staff therefore proposes to characterize the requested change as involving no significant hazards consideration.

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

Attoney for Licensee: Gerald Charnoff, Esquire, Jay E. Silberg, Esquire, Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Stevan A. Varga.

Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of amendment request: July 12, 1985.

Description of amendment request: Present surveillance requirements (Technical Specifications Section 4.3.3.3.2) specify recalibrating the seismic monitoring instruments within 24 hours following a seismic event. The manufactures states that a typical channel calibration would take a minimum of 5 days assuming that all of the instruments could be removed at once and that the required personnel were available. If any equipment or personnel-related delays occurred, this would take even longer. Therefore, because of the complexity of the seismic instrumentation, the 24-hour requirement is impractical.

It is important to calibrate seismic instrumentation soon after a seismic event in order to confirm instrument characteristics and validate data reduction. However, it is also important not to remove the entire system from operation immediately following a seismic event since this is the period of time during which aftershocks may occur. If all of the seismic instrumentation were taken off-line for a channel calibration immediately after a seismic event, important aftershock data would probably be missed.

Therefore, for the reasons given above, the licensee recommends revising the surveillance requirements from 24 hours to 30 days to allow sufficient time for both aftershock recording and channel calibration in phases. Calibration in phases is recommended to allow part of the seismic instrumentation to be on-line at all times.

Basis for proposed no significant hazards consideration determination: The proposed amendment does not involve any modification of existing instruments, nor does it alter the operational procedure of any equipment. While the amendment would allow more time to complete channel calibration, it also ensures that some instruments (those that are not being calibrated) are available to monitor aftershock data. We conclude that the proposed amendment would not invlove any significant increases in the probability or consequences of an accident previously evaluated, would not create the possibility of a new or different kind of accident from any accident previously analyzed, and would involve no reduction in the margin of safety. We, therefore, propose to characterize the proposed amendment as involving no significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esquire, Jay E. Silberg, Esquire, Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Stevan A. Varga.

Florida Power and Light Company, Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: July 19, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications to permit continued operation at rated thermal power for a specified time following a dropped control element assembly. Also, the current action statement C in Technical Specification 3.1.3.1 would be reformulated into new action statements C and H. This reformulation will better correlate the requirements for corrective action.

Basis for proposed no significant hazards consideration determination: The proposed changes would recognize the distinctions in safety analysis requirements by reconstructing the present action statement into two different action statements; one with applicability when control element assemblies are above the long term insertion limits and a separate one when the control element assemblies are inserted beyond the long term insertion limits. This separation will also aid in the standardization of technical specifications between St. Lucie Plant, Unit Nos. 1 and 2. No changes in safety analysis results or input are required as a result of this separation, or the addition of Figure 3.1-1a. Therefore, as required by 10 CFR 50.92(c)(1), the proposed changes do not result in an increase in the probability or consequences of any accident previously evaluated because no change in analysis input or assumptions is required for any transient.

The proposed changes to the technical specifications do not create the possibility of a new or different type of accident from any accident previously evaluated because neither the configuration of the plant nor its mode of operation will be modified. Therefore, there is no increase in the possibility of a new of different type of accident as discussed in 10 CFR 50.92(c)(2).

The proposed changes will not result in any reduction in the margin of safety as discussed in 10 CFR 50.92(c)(3) because no inputs to, or results from, plant safety analysis require change or modifications.

The required overpower margin for each transient analyzed for St. Lucie 1 is unaffected by the proposed changes, therefore, the difference between reactor safety limits and the results of the safety analysis, which is representative of the margin of safety, is unchanged.

Based on the above information, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, N.W., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

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

Date of amendment request: May 15 and May 21, 1984 as modified May 23 and August 26, 1985.

Description of amendment request: The proposed amendments would make changes in the technical specifications of St. Lucie Plant, unit Nos. 1 and 2 to reflect organizational changes, administrative changes (such as alphabetizing the definitions in the Unit 1 technical specifications) and changes to reflect the requirements of 10 CFR 50.72 and 50.73 as defined in Generic Letter 83-43.

The original changes that were proposed were contained in applications dated May 15 and May 21, 1984. The newest versions of the proposed amendments address NRC staff comments on the original applications and the staff's request for additional information dated December 7, 1984. The licensee's responses are contained in letters dated may 23 and August 26, 1985 and address the specific requests of the staff by enclosing modified technical specifications that update the organization charts and clarify compliance with 10 CFR 50.72 and 50.73.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards contained in 10 CFR 50.92(c) by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the **Federal Register** on April 6, 1983 (48 FR 14870). One of the examples of actions involving no significant hazards consideration, (i), relates to amendments of a purely administrative change to technical specifications, correction of an error, a change in nomenclature, or a change to achieve consistency throughout the technical specifications. The proposed changes to the St. Lucie Plant, Unit Nos. 1 and 2 Technical Specifications meet this example, in part, in that the amendments provide current organizational charts and amend the sections dealing with definitions and administrative controls to obtain consistency between the Unit 1 and Unit 2 Technical Specifications. Another example of actions involving no significant hazards consideration (vii) relates to changes to make a license conform to changes in the regulations, where the license changes result in very minor changes to facility operations clearly in keeping with the regulations. The proposed changes to the St. Lucie Plant, Unit Nos. 1 and 2 Technical Specifications meet this example, in part, in that the amendments incorporate the requirements of 10 CFR 50.72 and 50.73 as defined in Generic Letter 83-43. These changes deal exclusively and reporting requirements and do not affect plant operations.

Based on the above discussion, the staff concludes that the proposed changes will 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, because the changes do not affect the assumptions and inputs used in previously analyses nor do they affect in any way the normal operation of St. Lucie Plant, Unit Nos. 1 and 2. Therefore, the requirements of 10 CFR 50.92(c) are satisfied.

The NRC staff previously issued its proposed determination that the applications for amendments dated May 15 and May 21, 1984 did not involve a significant hazards consideration (June 20, 1984 at 49 FR 25359 for St. Lucie Plant, Unit No. 1 and July 24, 1984 at 49 FR 29909 for St. Lucie Plant, Unit No. 2).

Based on the above, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

Florida Power & Light Company, Docket No. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: July 19, 1985.

Description of amendment request: The proposed amendment for St. Lucie Plant, Unit No. 1 would amend operating license DPR-67 technical specifications to add Incore Thermocouples, Containment Sump Water Level (narrow and wide ranges), Containment Pressure, and Reactor Vessel Level Monitoring System to Tables 3.3-11 and 4.3-7. The amendment for St. Lucie Plant, Unit No. 2 would amend Operating License NPF-16 technical specifications to add the Reactor Vessel Level Monitoring System to Tables 3.3-10 and 4.3-7. Appropriate operability and action statements will be added, also.

Basis for proposed no significant hazards consideration determination: The requested changes to the Technical Specifications (TS) would revise the tables to add instrumentation that is currently installed and operational. These additions will provide a higher degree of control and operational readiness by requiring operability and monthly surveillance of accuracy.

The requested license amendments do not increase the probability or consequences of accidents previously analyzed. The plant hardware and normal operating conditions are not affected by the proposed changes. Addition of monthly surveillances will not involve any significant increase in the probability or consequences of an accident previously evaluated, but will assure accurate output from these instrument channels. Based on this, the criteria set forth in 10 CFR 50.92(c)(1) is satisfied.

The plant hardware and basis plant operation are not affected by the proposed changes. Therefore, the possibility for a new or different type of accident is not created and the criteria of 10 CFR 50.92(c)(2) is satified.

Since the consequences of accidents previously evaluated are not increased and no new or different types of accidents are introduced, all margins of safety will be maintained. This satisfies the criteria of 10 CFR 50.92(c)(3).

In addition, the Commission has provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the **Federal Register** on April 6, 1983 (48 FR 14870). One of the examples of actions involving no significant hazard consideration, (ii), relates to changes that constitute an additional limitation, restriction, or control not presently included in the technical specifications, for example, a more stringent surveillance requirement. The proposed changes provide for additional control and more stringent surveillance requirements and the above cited example directly applies to this application.

Based on the above considerations, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library 3209 Virginia Avenue, Fort Pierce, Florida 33450.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: August 31, 1984 as modified April 12, 1985.

Description of amendment request: The proposed amendment would make changes to the technical specifications that would limit the use of the 8-inch containment purge system to required safety related purposes, such as (1) maintaining containment pressure within the technical specification limit, and (2) reducing containment atmosphere activity and/or improving air quality to an acceptable level for containment entry to conduct safety related tasks.

The original application for amendment dated August 31, 1984 proposed that restrictions on the use of the 8-inch containment purge system be removed and that continuous purge using the 8-inch purge system be allowed. The staff found that continuous purge was unacceptable. The newest version of the proposed amendment addresses this staff position and was the result of several discussions with the licensee. The results are documented in the licensee's submittal of April 12, 1985. This submittal modifies the original application to limit the use of the 8-inch containment purge system to required safety related purposes, as indicated above.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a 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.

A discussion of these standards as they relate to this amendment follows:

Standard 1—Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed technical specification would allow operation of the containment purge system for safety related purposes. This represents a potential increase in operating time now limited to 1000 hours, or less, per year. This would not increase the probability of an accident since this system cannot, in itself, cause an accident. The system does serve to mitigate the consequences of a potential release to the public following a Loss of Coolant Accident (LOCA). In the evaluation of the system's valves, they are assumed to be open when a LOCA occurs. The valves are designed to close within 5 seconds of the start of a containment isolation actuation signal. This meets NRC Branch Technical Position CSB 6-4. Further, the system has been designed to accommodate a single failure. In the event of an accident, offsite doses will not exceed the limits specified in 10 CFR Part 100.

Standard 2—Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated

Allowing the use of the 8-inch containment purge system for safety related purposes does not involve any evolution that is not currently performed. In addition, the system, in itself, cannot cause an accident; therefore, operation for safety related purposes does not lead to the possibility of a new or different kind of accident from any previously evaluated.

Standard 3—Involve a Significant Reduction in a Margin of Safety

The containment purge system was originally designed for continuous operation. In the event of a LOCA, with a failure of a single 8-inch purge valve to close, the remaining valves will close within 5 seconds. Offsite doses due to a LOCA and one 8-inch purge valve failure will not exceed 10 CFR Part 100 limits.

Allowing purge through the 8-inch containment purge system for safety related purposes does not place the plant in a different configuration than what is currently routine practice. Therefore, the operation of the 8-inch purge system in this manner does not involve a significant reduction in a margin of safety.

The Commission has also provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the **Federal Register** on April 6, 1983 (48 FR 14870). One of the examples of actions involving no significant hazards consideration (iv) relates to a relief granted upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation was not yet demonstrated. This assumes that the operating restriction and the criteria to be applied to a request for relief have been established in a prior review and that it is justified in a safisfactory way that the criteria have been met. The proposed amendment, which would allow greater flexibility in the operation of the 8-inch purge system, is considered to be similar to example (iv) in that it involves relief from an operating restriction that was imposed prior to licensing because justification for the change requested in this amendment, based on plant operating experience, did not exist at that time.

The NRC staff previously issued a proposed determination that the original application for amendment dated August 31, 1984 did not involve a significant hazards consideration (September 28, 1984 at 49 FR 39390). The newest version of the proposed amendment is more restrictive than the original amendment application that was previously noticed.

Based on the above, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 33450.

Attorney for licensee: Harold F. Reis, Esquire, Newman and Holtzinger, 1615 L Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

General Electric Company, Docket No. 50-70, General Electric Test Reactor (GETR) Vallecitos Nuclear Center, Alameda County, California

Date of amendment request: June 26, 1985, as supplemented July 15, 1985.

Description of amendment request: By letter dated June 26, 1985, as supplemented July 15, 1985, GE requested an amendment to their license renewal application dated October 21, 1975 that would convert their current reactor operating license TR-1 to a possess-but-not-operate license. Prior notice of the application for renewal was given in **Federal Register** on September 15, 1977 at 42 FR 46427.

Basis for proposed no significant hazards consideration determination: All reactor fuel, fueled experiments, and targets containing SNM have been removed from the reactor facility and shipped from the Vallecitos Nuclear Center (VNC). In addition, all contaminated resins have been removed from the demineralizers and shipped to a licensed waste disposal facility. Therefore, only activation and fission product contamination remain. A confinement approach will be utilized to minimize the possibility of contamination spreads and uncontrolled discharges. The confinement system consists of primary containers, piping, the ventilation system, and the reactor building.

Activities to be performed at the facility would include decontamination testing and decommissioning training exercises. Work will be limited to equipment, components, or devices would or could be removed, repaired, replaced or installed as part of "normal maintenance" under the operating license. In addition, all such work would exclude installation or reinstallation of any fuel, equipment, component or device for the purpose of restoring the facility to a condition where it would be capable of operating as a nuclear reactor. The proposed activities would not involve the material alteration of the reactor facility.

The licensee has proposed Technical Specifications which (1) define the activities that require confinement and specify the actions to be taken and the equipment provided to achieve confinement; (2) provide assurance the reactor ventilation system is operable when required; (3) require that during the performance of restricted activities, the stack effluent shall be monitored or sampled; and (4) specify stack release rate limits that are lower than the current ones for the operating reactor facility.

There are no credible accidents which can be postulated which could result in the release of significant amounts of radioactive materials.

For the above reasons, the staff concludes that the proposed activities do 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.

Therefore, the staff proposes to determine that the requested action does not involve a significant hazards consideration.

Attorney for licensee: George Edgar, Esquire, Newman and Holtzinger, 1615 L Street, NW., Suite 1000, Washington, D.C. 20036.

NRC Branch Chief: Cecil O. Thomas.

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

Date of amendment request: June 24, 1985.

Description of amendment request: The amendment would add words that were inadvertently left out of Technical Specification 4.5.D.2 during retyping for a submittal dated April 22, 1983.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance for the application of the criteria in 10 CFR 50.92 by providing examples of amendments that are considered not likely to involve a significant hazards consideration (48 FR 14870). One such example (i) of action not likely to involve significant hazards consideration is a purely administrative change to the Technical Specifications. The change to the Technical Specifications described above is similar to this example.

On the basis of the above, the Commission has made a proposed determination that the application for amendment involves no significant hazards consideration.

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

Attorney for licensee: G.F. Trowbridge, Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: August 23, 1985.

Description of Amendment Request: The amendment would make changes in the license conditions and Technical Specifications necessary to modify and test one of two trains of the standby service water (SSW) system during the fall 1985 scheduled outage to satisfy, in part, License Condition 2.C.(20).

The changes proposed are: (1) Change Technical Specifications 3.8.1.1 and 3.8.1.2 to require diesel fuel storage for each of diesel generators 11 and 12 to be increased from 48,000 to 57,200 gallons to be consistent with larger SSW pumps to be installed; (2) change License Condition 2.C.(20) to allow valves isolating the spent fuel pool coolers to be opened provided the associated SSW subsystem is declared inoperable in accordance with Technical Specifications; and (3) add a license condition to allow a temporary exception to Technical Specification 3/4.7.1.3 (which requires a 30 day water supply in the SSW cooling tower basin without makeup) provided a specified water level is maintained and two sources of makeup in addition to normal makeup are available.

Basis for proposed no significant hazards consideration determination: The proposed changes are needed to allow modifications to the standby service water system (SSW) during the outage scheduled in October and November 1985. The License Condition 2.C(20) requires modifications to the SSW system and verification that design flow can be achieved in all SSW system components prior to storing irradiated fuel in the spent fuel pool and requires the spent fuel pool coolers to be isolated from the SSW system by locked closed valves until the modifications and verification tests are completed. The licensee is planning to modify Train B of the SSW system in the fall 1985 outage and will modify Train A during the first refueling outage. The modifications proposed for the fall 1985 outage include installation of a larger capacity SSW pump in SSW cooling tower Train B which requires draining of the basin. Modifications also include relocation of the SSW loop B supply and return valves to the spent fuel pool cooler which requires taking the spent fuel pool cooler in the B loop out of isolation. Verification tests of design flow to spent fuel pool coolers will also require taking the spent fuel pool cooler out of isolation. The modifications and tests will be made while the plant is in cold shutdown. Prior to startup following the 1985 outage, the SSW cooling tower basin will be refilled and the spent fuel pool coolers will be isolated until the other SSW loop modifications are completed.

The design change will be performed in accordance with appropriate regulatory and industry codes and standards, the GGNS Quality Assurance Program, and the applicable requirements of the GGNS FSAR. The proposed technical specification changes to the fuel capacities (change 1) will make the technical specifications consistent with the plant as modified by the proposed design change. The proposed revision to License Condition 2.C.(20) (change 2) will allow a necessary design change to prevent water hammer in SSW loop B piping, permit SSW loop B flow testing and permit evolutions involving alternate decay heat removal methods. Since the SSW subsystem associated with an open valve to the fuel pool cooler must be declared inoperable by the proposed change, assurance is provided by appropriate technical specification action statements that the plant will be maintained in a safe condition. The proposed temporary license condition (change 3) will be in effect only while the plant is in cold shutdown. The major heat load handled by SSW during cold shutdown is decay heat from the reactor fuel. After day 17 from plant shutdown the dominant heat load on the SSW system (after a loss of offsite power coincident with the design basis loss of coolant accident) is from running equipment instead of reactor decay heat. The probability of a design basis loss of coolant accident while the plant is in cold shutdown is low and the consquences of the accident are small in comparison to having the accident at 100% power. The diversity of water sources offered for makeup to SSW basin A will ensure a water supply after the present capacity of 16 days (from day eight of the outage) is used. Even in the unlikely event that no makeup can be provided, the inventory of water between 107' and 84' (providing an additional 16 day supply) can be utilized with a relatively small reduction in the design flow to equipment. If the provisions of the proposed license condition cannot be met, SSW basin A will be declared inoperable and appropriate technical specification action requirements will be implemented.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because of the low probability of an accident while the plant is in cold shutdown and the consequences of an accident in cold shutdown are not as severe as when the plant is at power. When the design change is complete, the increased flow and basin draindown ability of the new SSW pump will provide greater heat removal capacity than the present pump. Design requirements of 30 days without makeup water are assured (by use of the siphon between SSW basins A and B if required) until the first refueling outage. MP&L will provide a submittal prior to the first refueling outage to request operating license changes to facilitate SSW basin A design changes and removal of present restrictions on SSW operability and spent fuel storage prohibitions.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the license condition ensures an adequate supply of water to SSW basin A while the design change is in progress. Once the design change is completed, SSW basin design requirements of 30 days without makeup water will be assured (by use of the siphon if required) until the first refueling outage. MP&L will provide a submittal prior to the first refueling outage to request operating license changes to facilitate SSW basin A design changes and removal of present restrictions on SSW operability and spent fuel storage prohibitions. Thus, no new or different accident scenarios are postulated by performing the proposed design change to SSW basin B.

The proposed changes do not involve a significant reduction in a margin of safety because the proposed license condition ensures design water supply for SSW basin A while the design modifications are being performed and the return to the existing water supply when the modifications are completed and prior to a return to power operation.

Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 39154.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, NW., Washington, D.C. 20036.

NRC Branch Chief: Elinor G. Adensam.

Northeast Nuclear Energy Company, et al., Nos. 50-245 and 50-336, Millstone Nuclear Power Station, Unit Nos. 1 and 2, New London County, Connecticut

Date of amendment request: May 22, 1985 and supplemented by letter dated July 2, 1985.

Description of amendment request: The proposed amendments to the Operating Licenses would incorporate the proposed Revision 3 to the Suitability, Training & Qualification Plan. These amendments would: (1) Permit the use of contingency guard force personnel not normally assigned to the guard force to be assigned to replace striking guard force personnel; and (2) eliminate the requirement that Security Shift Supervisors (SSS) be required to requalify annually in crucial tasks performed by watchmen and guards, unless they are assigned to these tasks as a member of the contingency guard force. Any additional description of the changes beyond that stated above involves Safeguards Information, which is being withheld from public disclosure pursuant to 10 CFR 73.21.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a 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 use of contingency guard force personnel to replace striking guard force personnel to perform specific crucial tasks is acceptable in that all contingency guard replacement personnel will satisfy all suitability, physical and mental requirements of the Suitability Training and Qualification Plan prior to performing any tasks.

The elimination of the requirement that Security Shift Supervisors (SSS) be required to requalify annually in crucial tasks performed by watchmen and guards, unless they are assigned to these tasks as a member of the contingency guard force, is acceptable because: The SSS will continue to qualify in these tasks during their initial training, and the need to utilize SSS as watchmen or guards is remote because of the availability of fully trained guard force personnel, and if the need should arise, SSS could, and would, be trained to perform specific watchmen and guard duties without unreasonable delay.

Therefore, the proposed change would not involve or create any of the three factors quoted above. Accordingly, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Gerald Garfield, Esq., Day, Berry and Howard, One Constitution Plaza, Hartford, Connecticut 06103.

NRC Branch Chief: John A. Zwolinski (Unit 1) and Edward J. Butcher, Acting (Unit 2).

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

Date of amendment request: July 24, 1985.

Description of amendment request: The proposed change to the Technical Specifications would authorize the licensee to increase the spent fuel pool storage capacity from 667 to 1112 storage locations. The proposed expansion is to be achieved by reracking the spent fuel pool with a combination of poison racks and non-poison racks in a two-region arrangement.

Region I consists of two 8 × 9 modules and three 8 × 10 modules and would store high-enrichment, core off-load assemblies. The region consists of poisoned spent fuel racks with a nominal center-to-center cell spacing of 9.8 inches. Fuel assemblies would be stored in every location. The five modules of Region I total 384 storage locations and are designed to accommodate 1.7 reactor cores of high enrichment nuclear spent fuel.

The spent fuel rack design for Region I is based upon the commonly accepted physics principle of a "neutron flux trap" with the use of neutron absorber materials. The racks are designed to store Millstone 14 × 14 fuel with an initial enrichment of 4.5 weight percent U-235. The poison material to be used is Boraflex.

Region II consists of 14 modules of non-poisoned spent fuel racks with nominal center-to-center cell spacing of 9.0 inches. The modules consist of 962 cells with useable capacity of 728 storage locations.

Region II is reserved for fuel that has sustained at least 85% of its design burn-up. The spent fuel rack design is based on the criticality acceptance criteria specified in Revision 2 of Regulatory Guide 1.13 which allows credit for reactivity depletion in spent fuel. (Previously, the physics criteria for fuel stored in the spent fuel pool were defined by the maximum unirradiated initial enrichment of the fuel). Fuel assemblies are stored in a three-out-of-four logic pattern. The fourth location of the storage configuration remains empty to provide the flux trap to maintain the required reactivity control. Blocking devices will be used to prevent inadvertent placing of a fuel assembly in the fourth location.

The spent fuel racks in both regions are fabricated from 304 stainless steel which is 0.135 inches thick. Each cell is formed by welding along the intersecting seams. This enables each spent fuel rack module to become a free-standing module that meets the seismic design requirements without mechanical dependence on neighboring modules or fuel pool walls for support. The rack modules are classified ANS Safety Class III and Seismic Category I.

Both regions of the spent fuel have been designed to store fuel assemblies in a safe, coolable, subcritical configuration with K_{eff} less than or equal to 0.95.

The racks have been designed and will be provided by Combustion Engineering, Inc. (CE). CE racks of this type have been most recently licensed by the NRC for use at Florida Power and Light Company's St. Lucie Plant and at Arizona Public Services Company's Palo Verde nuclear plants. *Basis for proposed no significant hazards consideration determination:* The technical evaluation of whether or not an increased spent fuel pool storage capacity involves significant hazards considerations is centered on three standards.

A. First Standard

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

The licensee's safety analysis of the proposed reracking has been accomplished using current NRC Staff accepted Codes and Standards. The results of the safety analysis demonstrate that the proposal meets the specified acceptance criteria set forth in these standards. In addition, the licensee has reviewed NRC Staff Safety Evaluations (SEs) for prior spent fuel pool rerackings involving spent fuel pool rack replacements to ensure that there are no identified concerns not fully addressed. The licensee has identified no such concerns.

The licensee has identified the following potential accident scenarios: (1) Spent fuel cask drop; (2) loss of spent fuel pool forced cooling; (3) seismic event; (4) spent fuel assembly drop; (5) criticality accident; and (6) Load Handling Accident. The probability of the occurrence of any of the first four listed accidents is not affected by the racks themselves; thus, reracking cannot increase the probability of these accidents.

All potential events which could involve accidental critically have been examined in the licensee's safety analysis. It was concluded that the bounding accident was dropping an unirradiated fuel assembly into a blocked fourth location in Region II. The probability of dropping a fuel assembly during fuel movement operations is not affected by the fuel storage racks.

The proposed Millstone Unit No. 2 spent fuel pool reracking will not involve an increase in probability of any previously evaluated load handling accident as accepted standards and procedures will be utilized as described in the licensee's safety analysis.

The consequences of the spent fuel cask drop accident have been evaluated as described in Sections 5.4 and 9.8 of the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). By controlling the decay time for fuel stored within a specified distance from the cask set-down area to not less than 120 days prior to casks movement together with an administrative control specifying a minimum required boron concentration in the water of the spent fuel pool, the consequences of this accident type will remain well within 10 CFR Part 100 guidelines.

There is, however, an increase in the value of the 2-hour whole body dose at the site exclusion boundary for a postulated cask drop accident. The new racks increase the storage density of spent fuel within the distance L of the cask set-down area. This results in a calculated increase of the 2-hour whole body dose from 140 millirem to 240 millirem, an increase of 100 millirem. In review of this submittal, the licensee has recognized this increase and has designated it an unreviewed safety question. The calucated dose is well within the guidelines specified by 10 CFR Part 100 and, as such, the consequences of this type of accident will not be significantly increased from previously evaluated events.

The consequences of the loss of spent fuel pool forced cooling accident have been evaluated and are described in the licensee's safety analysis. There is ample time to effect repairs of the cooling system or to establish makeup flow to the spent fuel pool. The consequences of this type accident will not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a seismic event have been evaluated against the appropriate NRC standards. The results of the seismic and structural analysis show that the proposed racks meet all of the NRC structural acceptance criteria and are consistent with results found acceptable by the NRC Staff in previous poison rerack SEs. Thus, the consequences of seismic events will not significantly increase from previously evaluated seismic events.

The consequences of a spent fuel assembly drop accident are described in Section 14.19 of the Millstone Unit No. 2 FSAR. A complete list of assumptions is provided in FSAR Table 14.19-1. Results of the analysis are well below the limits of 10 CFR Part 100 and are presented in Section 14.19.3. The consequences of this type accident will not be significantly increased from previously evaluated accidents by this proposed reracking.

The consequences of a criticality accident have been evaluated for all potential events which could involve accidental criticality. The bounding criticality accident was found to be the dropping of a fresh fuel assembly into a blocked fourth location in Region II. Administrative controls in the form of a Technical Specification of minimum boron concentration for the water of the spent fuel pool will preclude the bounding criticality accident; therefore, the consequences of this type accident will not be significantly increased from previous accident evaluations by this proposed reracking.

The consequences of a load handling accident have been evaluated. The work to be done in the spent fuel pool will be performed in accordance with accepted construction practices, standards, and procedures. The consequences of this type accident will not be significantly increased from previous accident evaluations by this proposed reracking. Therefore, it is shown that the proposed Millstone Unit No. 2 spent fuel rack replacement will not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. Second Standard

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

The licensee has evaluated the proposed rack replacement in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plan sections, and appropriate industry Codes and Standards. In addition, the licensee has reviewed the NRC Safety Evaluation for the previous Millstone Unit No. 2 spent fuel rack replacement application and for other prior spent fuel pool rerackings.

The change to a two-region spent fuel pool creates the requirement to perform additional evaluations to ensure the criticality requirement is maintained. These include the evaluation of the limiting condition (dropping a fresh fuel assembly into a blocked fourth location in Region II). This evaluation shows that, when the boron concentration requirement is met per the proposed Technical Specifications, the criticality criterion is satisfied. Although this change does create the requirement to address additional aspects of a previously analyzed accident, it does not create the possibility of a previously unanalyzed accident.

C. Third Standard

Involve a significant reduction in a margin of safety.

The issue of "margin of safety," when applied to a spent fuel rack replacement, includes the following considerations:

a. Nuclear criticality considerations.

b. Thermal hydraulic considerations.

c. Mechanical, material, and structural considerations.

The margin of safety that has been established for nuclear criticality is that the neutron multipliciation factor (K_{eff}) in the spent fuel pool is to be less than or equal to 0.95, including all uncertainties, under all conditions. For the proposed modification, the criticality analysis is described in the licensee's safety analysis. The methods utilized in the analysis conform with ANSI N210-1976, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations"; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" (April 1978), as modified (January 1976); and Regulatory Guide 1.13, "Spent Fuel Facility Design Basis," proposed Revision 2. The computer programs, data libraries, and benchmarking data used in the evaluation have been used in previous spent fuel rack replacement applications by other NRC licensees and have been reviewed and approved by the NRC. The results of the licensee's analysis indicate that K_{eff} is less than or equal to 0.95 under all postulated conditions, including uncertainties, at a 95/95 probability/confidence level. Thus, meeting the acceptance criteria for criticality, the proposed reracking does not involve a significant reduction in the margin of safety for nuclear criticality.

For thermal hydraulics, the relevant considerations for evaluating if there is a significant reduction in margin of safety are: (1) Maximum fuel temperature, and (2) the increase in temperature of the water in the pool. The licensee's thermal hydraulic evaluation shows that fuel cladding temperatures under abnormal conditions are sufficiently low to preclude structural failure and that boiling does not occur in the water channels between the fuel assemblies nor within the storage cells. However, the proposed rack replacement will result in an increase in the maximum heat load in the Millstone Unit No. 2 spent fuel pool. The licensee's safety analysis shows that the maximum temperature will not exceed the current margin of safety (150°F). For the maximum normal heat load case (full-core discharge at 150 hr. after shutdown, which fills the spent fuel pool to its capacity), the pool temperature will not exceed 150°F. Thus, there is no significant reduction in the margin of safety from a thermal hydraulic standpoint or from a spent fuel pool cooling standpoint.

The mechanical, material, and structural considerations of the proposed rack replacement are also analyzed in the licensee's safety analysis. The racks are designed in accordance with the applicable NRC Regulatory Guides, Standard Review Plan sections, and position papers, and appropriate industry Codes and Standards, as well as to Seismic Category I requirements. The materials utilized are compatible with the spent fuel pool and the spent fuel assemblies. The conclusion of the analysis is that the margin of safety is not significantly reduced by the proposed reracking.

Based on the above discussion, the staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Gerald Garfield, Esq., Day, Berry and Howard, One Constitution Plaza, Hartford, Connecticut 06103.

NRC Branch Chief: Edward J. Butcher, Acting.

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

Date of application of amendment: June 27, 1985.

Description of amendment request: The proposed amendment will change the Technical Specifications (TS) to incorporate the changes as a result of implementation of NUREG-0737, Item II.K.3.16 requirements. The changes are as follows:

(1) Change the Safety/Relief valve self-actuation setpoint specified in Section 2.4.B from 1108 psig to 1120 psig.

(2) In Table 3.2.7, increase the Low-Low Set Logic opening and closing setpoints for Reactor Coolant System Pressure by 12 psi.

Basis for proposed no significant hazards consideration determination: NUREG-0737, Item II.K.3.16 required BWR licensees and BWR operating license applciants to investigate the feasibility of a number of actions and modifications to reduce challenges to SRVs. At the time, the operating history of SRVs had been poor, resulting in a relatively high failure rate per challenge. The evaluation was performed by the BWR Owners Group (BWROG-8134). The staff reviewed the BRW Owners Group study and endorsed three specific modifications along with an effective preventative maintenance program. The changes requested by this amendment (increasing valve simmer margin) is one of the staff approved modifications. This change will increase the self-actuation and Low-Low Set Logic setpoints of the safety/relief valves by 12 psi. This increase is safety/relief value simmer margin will increase valve reliability by reducing the probability of valve leakage and spurious opening during plant operation.

General Electric has performed the analysis supporting the 12 psi increase in safety/relief value settings. The results are provided in GE report NEDO-30771, dated September 1984. The analysis concludes that the setpoint increase is clearly within all acceptance criteria established in the NRC staff guidance. The staff, therefore, proposes that the changes 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.

Therefore, based on the reasons as described above, the staff has made a proposed determination that the application involves no significant hazards consideration.

Local Public Document Room location: Environment Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Cheif: Domenic B. Vassallo.

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

Date of amendment request: July 11, 1985.

Description of amendment request: The proposed amendment would revise the surveillance capsule removal schedule technical specifications, change the titles of senior management officials in the technical specifications, and delete environmental qualification of electrical equipment administrative requirements (deadline and central records location) from the technical specifications.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples, (i), of actions not likely to involve a significant hazards consideration relates to a purely administrative change to the Technical Specifications such as a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. The changing of the titles of senior management officials comes under this example because only the titles are changing and not personnel or functions. Another one of the examples, (vii), of actions not likely to involve a significant hazards consideration relates to a change which would make a license conform to changes in the regulations, where the change results in very minor changes clearly in keeping with the regulations. The deletion of environmental qualification (EQ) of electrical equipment administrative requirements from the technical specifications comes under this example because the EQ rule (10 CFR 50.49) now addresses the administrative requirements.

Regarding the revision of the capsule removal schedule, the licensee has made a significant hazards consideration determination pursuant to 10 CFR 50.92. The licensee has stated that (1) the proposed change will not result in an increase in the probability or consequence of a previously evaluated accident, but instead will provide better information on the fluence to the inside surface of the reactor vessel; (2) the proposed capsule surveillance schedule will not create the possibility of a new or different kind of accident from any previously evaluated accident because the capsule assemblies have not changed, only their sequence of removal; and (3) there is not change in any margin of safety involved in this Technical Specification change.

Based upon the above discussion, the staff proposes to determine that the proposed changes do not involve significant hazards considerations.

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

Attorney for licensee: LeBoeuf, Lamb, Leiby, and MacRae, 1333 New Hampshire Avenue NW., Washington, D.C. 20036.

NRC Branch Chief: Edward J. Butcher, Acting.

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

Date of amendment request: August 6, 1985.

Description of amendment request: The requested amendment would change the Technical Specifications for Susquehanna SES, Unit 1 to correspond with certain proposed design changes to the nitrogen makeup system.

In Licensee Event Report No. 83–114, dated September 13, 1983, PP&L noticed the NRC staff of the discovery of a postulated single failure event in the Division II Primary Containment Isolation System (PCIS) logic that could have resulted in the failure to isolate the nitrogen supply line. The PCIS Division II relay provides a closure signal to the outboard isolation valve of the drywell nitrogen supply system and the inboard isolation valve of the containment atmosphere control sample system. The drywell nitrogen supply line taps into the containment atmosphere sample line between the inboard valve and the outboard valve. With the nitrogen makeup system in service, coincident with a loss of coolant accident (LOCA), the PCIS Division II relay could fail in such a manner as to maintain the outboard isolation valve of the drywell nitrogen supply system and the inboard isolation valve of the containment atmosphere control sample system in the open position. This configuration could creat a direct path from the primary containment to the outside environment given the postulated single failure concurrent with a LOCA. A similar scenario can be postulated for the isolation valve in the suppression chamber nitrogen supply system and the inboard isolation valve for the containment atmosphere return line.

The design changes to correct this deficiency will consist of rerouting the drywell and wetwell makeup lines to spare penetrations and installing divisionalized isolation valves. The inboard valves will have Division I power and logic, and the outboard valves will be with Division II.

The licensee has proposed changes to Table 3.6.3–1 to ensure that the Technical Specifications properly reflect the installation of the modifications to the nitrogen makeup system. Those changes include the addition of two new isolation valves, SV–15738 and SV–15789. Two valves currently listed under the Containment Atmosphere Sample category, SV–15737 and SV–15767, are proposed to be deleted and moved to the newly formed category "Nitrogen Makeup", since in the new configuration, they will no longer be in the atmosphere sampling lines. The isolation signals for this new category are "B", Reactor Vessel Water Level—Low, Low Level 2; "Y", Drywell Pressure—High; and "R", SGTS Exhaust Radiation—High.

Valves SV–15736B and SV–15776B will now be dedicated to the sampling lines. Therefore the "R" isolation signal, SGTS Exhaust Radiation-High, is no longer applicable and is being deleted from the Technical Specifications for these valves.

Basis for Proposed No Significant Safety Hazards Consideration Determination

The licensee has stated that:

I. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The addition of the nitrogen makeup line is compatible with the FSAR design requirements. The proposed action does not increase the probability of an occurence or consequence of an accident or malfunction of equipment related to safety. All engineering has been performed in accordance with plant design criteria to assure the required installation will not impact safety-related systems. The results of the most recent integrated leak rate test will be adjusted based on local leak rate testing of the new nitrogen makeup system configuration, when it is installed. Bypass leakage effects discussed in FSAR Subsections 6.2.3 and 6.2.1.1.5 remain unchanged by this proposed action.

II. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Design of the proposed new penetrations, isolation valves and isolation circuitry is in accordance with the existing design basis of the plant.

III. The proposed change does not result in a significant reduction in a margin of safety. This change will increase the safety margin of the plant by ensuring that a single failure in the nitrogen makeup system isolation logic will not allow an uncontrolled release of radiation to the environment following a design basis accident. As stated in II above, the design change will be in accordance with the existing design basis.

The NRC staff agrees with the licensee's evaluation in this regard and proposes to find the proposed change to not involve a significant hazards consideration based on the licensee's ability to meet the three criteria described above.

Identical modifications have already been made for SSES Unit 2, with the corresponding Technical Specifications revised via Amendment 2 to License NPF-22, dated October 9, 1984.

Local Public Document Room Location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Attorney for Licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 1800 M Street NW., Washington, D.C. 20036.

NRC Branch Chief: Walter R. Butler.

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

Date of amendment request: April 19, 1985.

Brief description of amendment: The application requests modifications to the Technical Specifications to: (1) Revise the definition of Operability to provide the additional guidance that only items necessary for a system to perform its safety-related function are necessary to demonstrate operability, and revise the title to Definition 1.4 to include the word MODE; (2) revise the Action statement associated with the structural integrity of the RCS to define what must be done when the structural integrity of a Class 1 or 2 component is discovered to be questionable when the plant is already over the temperature limits given in the Technical Specifications, and place restrictions on the time allowed for evaluation and temporary repair; and (3) revise the surveillance requirements of the Control Room Emergency Ventilation System (CREVS) to require that testing done to verify that the CREVS can maintain design control room temperatures only be performed once a year rather than every 31 days.

Basis for proposed on significant hazards consideration determination: The licensee has provided a discussion of the proposed amendment with respect to the issue of no significant hazards consideration which is presented below for each issue.

1. *Definition of Operable—Operability.*

This revision clarifies the definition by stating that non-safety-related portions of systems are not necessary to declare a safety-related system OPERABLE. The definition continues to require that all safety-related systems, subsystems, trains, components, or devices will be able to perform their safety-related function in order to be declared OPERABLE. Therefore, this revision does not impose a significant hazard consideration.

The staff has reviewed this discussion presented by the licensee and agrees that the proposed amendment is meant to clarify without degrading the definition of operability. The licensee has also proposed to add the word MODE to the title of Definition 1.4. The Commission has provided guidance to the NRC staff for a proposed no significant hazards consideration determination by providing examples of amendments that are not likely to involve a significant hazards consideration. One example is (i), a purely administrative change to technical specifications. The licensee's proposed change to the definition of operability, which is a clarification of the definition, and the revision to Definition 1.4's title fall under the domain of this example and therefore do not involve a significant hazards consideration.

2. *Structural Integrity.*

The present wording for ACTION Statement d does not specify a time limit for the evaluation or repair allowed by the ACTION statement. The statement has been revised to specify that the evaluation must be completed within 72 hours and repairs within the following 36 hours. The 72- and 36-hour limits are not based upon a specific accident analysis but upon a compromise between the time needed to perform the work and the need to ensure that the plant is not operated for an extended period with potentially degraded structural integrity. Since the revision prevents extended operation with potentially degraded structural integrity, a significant hazard consideration is not deemed to exist for this change.

The staff has reviewed the licensee's determination of no significant hazards consideration for this change to the Technical Specification and proposes to agree with that determination.

The proposed change is meant to clarify what must be done when the structural integrity of a Class 1 or 2 component is discovered to be questionable when the plant is already above the temperature limits given in the Technical Specification Action statements. This change is encompassed by example (i), as discussed above, of actions not likely to involve significant hazards considerations. Additionally, the licensee has proposed constraints on the time allowed for evaluation and repair of degraded components. This falls under example (ii) of actions not likely to involve a significant hazards consideration; that is, a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications.

3. *Control Room Emergency Ventilation System (CREVS).*

Survellance Requirement 4.7.6.1.a presently requires a test of each train of the control room emergency ventilation system every 31 days. This test performs two major functions:

a. Maintains the charcoal absorbers in a dry condition to ensure that they will be able to fulfill their design function, and

b. Verifies that the service water coolers can perform their function of maintaining the control room temperature within design requirements.

The revised Surveillance Requirement 4.7.6.1.a will perform testing to meet Item a above every 31 days as is presently done. However, Item b above will now be performed annually per Surveillance Requirement 4.7.6.1.b instead of the present interval of 31 days. This increased surveillance interval is considered acceptable based on the following:

a. The emergency ventilation system has never failed to maintain temperature within design conditions during all testing performed to date, and

b. Modes of possible failure of the service water coolers, such as fouling, do not occur rapidly enough to justify testing on a monthly basis. Annual testing is considered adequate to detect any decrease in the capacity of the service water coolers.

Based upon the above, a significant hazard consideration is not deemed to exist.

The staff has reviewed this discussion presented by the licensee. It appears that operation of the facility in accordance with the proposed changes would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated because testing will still be performed to demonstrate the capability of the CREVS to maintain the control room design basis temperature. The testing will be performed on a less frequent basis, but this should not impact the probability or consequences of an accident; or (2) create the possibility of a new or different kind of accident because a safety limit or limiting condition for operation has not been modified; or (3) involve a significant reduction in a margin of safety because annual testing appears adequate to detect any changes in the capacity of the service water coolers, and this test frequency is consistent with the rate at which degradation may occur.

Based on the foregoing, the NRC staff proposes to determine that the proposed amendment does not involve a significant hazards consideration.

Local Public Document Room location: Multnomah County Library, 801 SW. 10th Avenue, Portland, Oregon.

Attorney for licensee: J. W. Durham, Senior Vice President, Portland General Electric Company, 121 SW. Salmon Street, Portland, Oregon 97204.

NRC Branch Chief: Edward J. Butcher, Acting.

Sacramento Municipal Utility District, Docket No. 50-312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of amendment request: June 6, 1983, as supplemented June 29, 1983, April 3, 1984, July 11, 1984, November 28, 1984, February 8, 1985, and April 3, 1985.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to:

1. Change the requirement for calibration of the nuclear instrumentation power range amplifier from whenever indicated neutron power and core thermal power differ by more than 2% to whenever the Nuclear Instrumentation indication is 10% above the core thermal power or 2% below thermal power.
2. Increase the frequency of performing a heat balance check from daily to once per shift.
3. Delete the present requirement for daily calibration during non-steady-state operation.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility 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; (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 licensee states that the calibration change (item 1) is needed because power range nuclear instruments (NI) that are within calibration limits at or near full power will read high (with respect to thermal power) at reduced power levels (a conservative error). However, if the NI are brought within calibration limits at reduced power, they will then read low (a non-conservative error) when the reactor returns to full power. This condition will then exist until the next calibration. To remove this temporary non-conservative condition, the licensee proposed a change that will require calibration of the power range nuclear instruments when the heat balance exceeds neutron indicated power by 2%.

With regard to item 2, the licensee noted that power imbalance and control rod insertion limits become more restrictive as power level increases. Therefore, neutron channels which are high impose stricter and more conservative limits on reactor operation. However, for human factors consideration related to unbounded out-of-calibration conditions, the licensee proposed an upper limit to 10% for nuclear indicated power exceeding heat balance.

With regard to item 3, the licensee states that the proposed TS changes would make the requirements for daily calibration of the power range neutron instrumentation during non-steady-state operation unnecessary. The reasons are that the proposed new Specification would require a heat balance check (comparison of indicated neutron power and core thermal power) at least once a shift and calibration whenever the nuclear instrumentation indication was 2% below thermal power or 10% above core thermal power. Since these requirements address any calibrations needed during steady-state or non-steady-state conditions more frequently than the present requirements, the present requirement can be deleted.

Since the proposed TS changes would (a) remove a calibration when performed at lower than full power level which results in a non-conservative condition at full power, (b) provide an upper bound limit for human factors consideration, and (c) delete a calibration that has been replaced by a more frequent calibration, the Commission's staff concludes that the proposed change 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. Therefore, the staff proposes to determine that the proposed amendment does not involve a significant hazards consideration.

Local Public Document Room location: Sacramento City-County Library, 828 I Street, Sacramento, California.

Attorney for licensee: David S. Kaplan, Sacramento Municipal Utility District, 6201 S Street, P.O. Box 15830, Sacramento, California 95813.

NRC Branch Chief: John R. Stolz.

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362; San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of Amendment Request: May 9, 1985 (Reference PCN-184).

Description of Amendment Request: The proposed change would revise Technical Specification (T.S.) ¾.10.1, "Special Test Exceptions —Shutdown Margin." T.S. 3.10.1 allows the shutdown margin to be reduced to less than the normal operating shutdown margin requirements during the performance of low power physics tests, provided that certain conditions are met. As one of these conditions, Surveillance Requirement 4.10.1.2 requires that all control element assemblies (CEA's) not fully inserted in the core be demonstrated to be capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing shutdown margin to less than the normal operating requirements. The proposed change will allow this surveillance to be performed within seven days prior to the tests instead of within 24 hours prior to the tests. This will enable low power physics testing to be completed without an additional trip to verify CEA insertability.

Low power physics tests are performed to verify core physics predictions. One of the test sequences measures CEA worths and may involve the reduction of shutdown margin as permitted by T.S. 3.10.1. Prior to initial criticality for performamce of the low power physics tests, rod drop testing is performed to demonstrate CEA insertability. The reactor is brought critical and stabilized at the test plateau (approximately 10^{-2}% power). The preferred sequence for low power physics testing has CEA worth measurements made last. Since approximately five days would have elapsed from when the hot rod drop tests were last performed, the reactor would have to be tripped again to demonstrate CEA insertion capability and satisfy the current 24 hour criteria. The proposed change would eliminate the necessity for an additional trip during low power physics testing by requiring CEA insertability to be verified within seven days prior to reducing shutdown margin instead of within 24 hours.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example a change resulting from the application of a small refinement of a previously used calculational model or design method.

In this case, SRP Section 14.2, "Initial Test Program" and SRP Sections 15.1.1, 15.1.2, 15.1.3, 15.1.4 and 15.1.5 which relate to Reactor Coolant System (RCS) overcooling events provide the pertinent acceptance criteria. SRP Section 14.2 refers to Regulatory Guide (R.G.) 1.68, "Initial Test Programs for Water Cooled Nuclear Power Plants." R.G. 1.68 outlines the elements of an acceptable startup test program including requirements for CEA worth measurements during low power physics testing. The proposed change will facilitate CEA worth measurements and is consistent with R.G. 1.68 and SRP Section 14.2.

The proposed change does not affect the consequences of RCS overcooling events evaluated in accordance with SRP Section 15.1.1 through 15.1.5. Because of a negative moderator temperature coefficient, RCS overcooling results in a reactivity increase. Because of this, a post trip return to power may be experienced in overcooling events if insufficient negative reactivity is inserted via the CEA's. Since shutdown margin is reduced during CEA worth measurements, T.S. 4.10.1.2 provides added assurance that all CEA's are trippable. By increasing the period during which shutdown margin may be reduced following performance of surveillance requirement 4.10.1.2, the proposed change may result in an insignificant reduction in the assurance provided. The resultant increase in the probability of a stuck CEA coincident with an overcooling event has been calcuated by the licensee to be 1.8×10^{-7}. The proposed change has no effect on the consequences of overcooling events since it does not affect the amount by which shutdown margin may be reduced. Because the consequence of these events are not increased, the SRP acceptance criteria continue to be satisfied. Based on these considerations, the NRC staff proposes to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room Location: San Clemente Library, 242 Avenida Del Mar, San Clemente, California 92672.

Attorney for licensee: Charles R. Kocher, Esq., Southern California Edison Company, 2244 Walnut Grove Avenue, P.O. Box 800, Rosemead, California 91770 and Orrick, Herrington & Sutcliffe, Attn.: David R. Pigott, Esq., 600 Montgomery Street, San Francisco, California 94111.

NRC Branch Chief: George W. Knighton.

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

Date of amendment request: July 17, 1984, as supplemented by letter dated October 3, 1984.

Description of amendment request: The proposed amendment would revise Technical Specification 4.6.1.2 and its associated bases regarding containment leakage surveillance requirements to proved clarifications on the leak rate testing of valves pressurized with fluid from a seal system. The clarifications are provided by the incorporation of Standard Technical Specifications 4.6.1.2.d.3) and 4.6.1.2.g (NUREG-0452, Revision 4) into Callaway Specifications 4.6.1.2.d.3) and 4.6.1.2.h and by an addition to bases 3/4.6.1.2 to include Callaway Plant specific requirements that are consistent with 10 CFR Part 50, Appendix J, Paragraph III.C.3.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). This amendment request is similar to the example of an action involving no significant hazards consideration which relates to a change to make the license conform to regulations, where the license amendment results in changes to facility operations clearly in keeping with the regulations.

Local Public Document Room locations: Fulton City Library, 709 Market Street, Fulton, Missouri 65251 and the Olin Library of Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Branch Chief: B. J. Youngblood.

Yankee Atomic Electric Company, Docket No. 50-29, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: July 19, 1983, as modified April 13, 1984 and August 16, 1984 and supplemented April 5, 1985.

Description of amendment request: The original proposed change to the Technical Specifications (TS) was noticed in the **Federal Register** February 24, 1984 (49 FR 7051). The supplemented proposed change modified some portions of the original proposed change as well as adding new proposed changes to the TS. The following items have been modified from the original proposed changes: (1) The licensee withdrew the request to allow the inner door lock to remain open when containment is occupied, and (2) withdrew the request to delete or add valves related to the low pressure surge tank, the secondary systems isolation valves, the component cooling safety valve discharge, and the purification pump connection.

The supplement to the proposed change would revise the TS to (1) add a torque testing requirement for all threaded pipe caps or threaded plugs used to provide containment integrity, (2) administratively delete containment isolation valves that were either physically removed from systems or repiped such that the valves are no longer required for containment isolation, (3) administratively add valves and blank flanges that, due to redesignation, reconfiguration of piping, addition of valves not previously identified in TS, or installation of new valves, which are required for containment isolation, (4) add valves, with notations to allow operation of certain valves to allow for surveillance testing, operation of component cooling water, and for sampling of containment atmosphere during postulated accident conditions, and (5) add new categories of containment isolation valves, in addition to the categories contained in the original proposed change, for Safety and Secondary Isolation Valves.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards by providing certain examples (April 6, 1983, 48 FR 14870) of amendments not likely to involve significant hazards consideration. Example (ii) involves a change that constitutes an additional limitation, restriction, or control not presently included in the TS.

Revision to the TS items (1) and (3) above are encompassed by this example. Item (1) adds a torque requirement for the pipe caps and threaded plugs that were proposed to be added to the TS in the original TS change request. The torque test requirement does not currently exist in the TS for these fittings. Item (3) proposes to add valves and blank flanges to the TS listing of containment valves that require testing in accordance with Appendix J to 10 CFR Part 50. These valves and blank flanges are part of plant modifications that removed the low pressure surge tank (LPST), connections to the purification pumps and the outside air particulate monitor (OAPM) from the containment boundary. The listing of the added valves and blank flanges is now provided for the new containment isolation valves that replaced valves removed or redesignated during the LPST, the purification pumps, and the OAPM modifications. Additionally, item (3) proposes to add valves and taps that were installed to allow connection of the hydrogen recombiner.

The staff has reviewed Items (2), (4), and (5) of the licensee's submittal in accordance with the standards of 10 CFR 50.92 and has determined that should these revisions be implemented, they 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 previoulsly evaluated, or (3) involve a significant reduction in a margin to safety. The basis for this determination follows:

Various containment isolation valves and check valves associated with the LPST, purification pump, and OAPM have been replaced as containment barriers by the modifications discussed in item (3). Item (2) proposes to administratively remove the replaced boundaries from the TS. Item (4) proposes to add notation to (a) allow the OAPM isolation valves to be opened to allow sampling of containment air, (b) allow operation of normally open component cooling water (CCW) isolation valves under administrative controls to provide for isolation when CCW is determined to be involved in an accident, and (c) allow for surveillance of the hydrogen recombiner inlet taps under the in-service inspection program.

The original request also proposed the reorganization of the containment isolation valve listing into nine new categories. Revision to the TS item (5) above proposes to add the following additional categories to Table 3.6-1:

F. Safety Valves (Subject to Type C testing)

G.1 Secondary Automatic Isolation (not subject to Appendix J)

G.2 Secondary Manual Isolation (not subject to Appendix J)

G.3 Secondary Remote Manual Isolation (not subject to Appendix J)

G.4 Secondary Safety Valves (not subject to Appendix J)

These additional categories were proposed to (a) provide for containment isolation barriers in lines that were modified in conjunction with the removal of the LPST as part of the containment boundary, and (b) provide a listing of main steam and feedwater systems isolation valves in the TS. One additional valve is being reclassified from the original proposed change as requiring type C testing. These changes to the original proposed change reflect the hardware modifications approved by the NRC to remove the LPST from the containment boundary, and provides a listing of the closed system isolation valves that are required by General Design Criterion 57.

Based on the above discussions, the staff proposes to conclude that none of the requested actions would involve a significant hazards consideration.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, 225 Franklin Street, Boston, Massachusetts 02110.

NRC Branch Chief: John A. Zwolinski.

Yankee Atomic Electric Company, Docket No. 50-29, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: June 26, 1985.

Description of amendment request: The proposed change would modify the surveillance interval requirements for Low and High Pressure Safety Injection Flow Tests, and for the Hot Leg Injection Flow Test, in the Technical Specification (TS). Additionally, the proposed change would renumber TS consistent with changes proposed to modify these surveillance intervals, and would correct a spelling error in the TS.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (April 6, 1983, 48 FR 14870). Example (i) of actions not likely to involve a significant hazards consideration determination involves a purely administrative change to the TS; for example, a change to achieve consistency throughout the TS, correction of an error, or a change in nomenclature. The proposed correction of a spelling error, and the proposed renumbering of the TS are consistent with this example.

The amendment request also proposes to change the Low Pressure Safety Injection Flow Tests from an 18 month surveillance interval, and the High Pressure Safety Injection and Hot Leg Injection Flow Test surveillance interval from 36 months, to an interval that requires these flow tests after completion of modification to ECCS subsystems that alter the subsystem flow characteristics. This proposed modification would make these flow test intervals for ECCS systems consistent with the Westinghouse Standard TS. The proposed change would, therefore, (1) not involve any significant increase in the probability or consequences of an accident previously evaluated; (2) not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) not involve a significant reduction in a margin of safety.

On these basis, the staff proposes to determine that the requested action would not involve a significant hazards consideration.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, 225 Franklin Street, Boston, Massachusetts 02110.

NRC Branch Cheif: John A. Zwolinski.

PREVIOUSLY PUBLISHED NOTICES OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO OPERATING LICENSES AND PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

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

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

Indiana and Michigan Electric Company, Docket No. 50-315, Donald C. Cook Plant, Unit No. 1, Berrien County, Michigan

Date of amendment request: July 30, 1985.

Brief description of amendment: Revise the Technical Specifications to reflect revised setpoints in the channels for overpressure delta T, overtemperature delta T, and loss of flow trips and the reactor coolant temperature to protect against departure from nucleate boiling (DNB).

Date of publication of individual notice in **Federal Register:** August 2, 1985 (50 FR 31447).

Expiration date of individual notice: August 16, 1985, 4:30 p.m.

Local Public Document Room location: Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

During the period since publication of the last bi-weekly 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 and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the **Federal Register** as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

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 amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C., and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

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

Date of application for amendments: January 31, 1985.

Brief description of amendments: The amendments changed the Unit 1 and Unit 2 Technical Specifications (TS) to reflect clarification and increased flexibility for determination of reactor coolant system leakage as specified in TS 3/4.4.6.1, "Leakage Detection Systems" and TS 3/4.4.6.2, "Reactor Coolant System Leakage."

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment Nos.: 107 and 88.

Facility Operating License Nos. DPR-53 and DPR-69. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 23, 1985 (50 FR 15997 at 15998).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

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

Baltimore Gas & Electric Company, Docket No. 50-318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of applications for amendments: February 26 and April 10, 1985.

Brief description of amendment: The amendment changes the Unit 2 Technical Specifications (TS) to reflect (1) analyses performed in support of Unit 1 Cycle 8 operation which is also applicable to Unit 2 which would allow more flexible limits for high pressure safety injection system flow, and (2) an increase from 24 hours to 7 days for the time period within which a scram test must be performed prior to reducing the shutdown margin below specified limits.

Date of issuance: August 30, 1985.

Effective date: August 30, 1985.

Amendment No.: 89.

Facility Operating License No. DPR-69. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 19, 1985 (50 FR 25480 at 25481).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 30, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 30, 1985.

Brief description of amendment: The amendment revises Technical Specifications Table 3.5-1 Items 6.a and 6.b to increase the voltage setpoint tolerances for loss of voltage and degraded grid voltage relays and increase the loss of voltage relay trip time. This completes our review of this item (TAC No. 57738).

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment No.: 93.

Facility Operating License No. DPR-74. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 19, 1985 (50 FR 25484).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29535.

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station, Unit Nos. 1 and 2, Benton County, Illinois

Date of application for amendments: January 30, 1985; supplemented July 8, 1985.

Brief description of amendments: These amendments would: (a) Permit full pressure testing of the containment in accordance with 10 CFR Part 50, Appendix J as requested by NRC-Region III; (b) add containment air lock testing; (c) reduce the number of tendons to be included in surveillance requirements in accordance with Regulatory Guide 1.35, Revision 2; and (d) remove obsolete containment liner surveillance requirements.

Date of issuance: August 27, 1985.

Effective date: August 27, 1985.

Amendment Nos.: 90 and 80.

Facility Operating License Nos. DPR-39 and DPR-48. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 23, 1985 (50 FR 16001).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 27, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Zion Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

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

Date of application for amendment: January 7, 1985 as revised March 14, 1985 which supersede previous requests dated June 4, 1976 and November 13, 1978.

Brief description of amendment: The amendment changes the Technical Specifications and Bases to incorporate Radiological Effluent Technical Specifications (RETS) which meet the requirements of Appendix I to 10 CFR Part 50. The amendment also includes administrative changes which relocate and reformat related Technical Specifications which are not part of the RETS, but were necessary in order to incorporate the RETS into the existing Technical Specifications.

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment No. 77.

Facility Operating License No. DPR-6. This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1985 (50 FR 23546).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Notrh Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

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

Date of application for amendment: June 13, 1984.

Brief description of amendment: The amendment issues changes to the technical specifications to (1) add limiting conditions for operation to require the containment purge and ventilation isolation valves to be electrically locked closed whenever the reactor is in a Hot Shutdown, Hot Standby, or Power Operation condition, and (2) add surveillance requirements to periodically check that these valves are closed and to periodically perform leak rate tests of the valves.

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment No. 90.

Provisional Operating License No. DPR-20. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 22, 1984 (49 FR 33362).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Van Zoeren Library, Hope College, Holland, Michigan 49423.

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

Date of application for amendment: June 15, 1985.

Brief description of amendment: The amendment changes the Technical Specifications to provide new, more restrictive pressure-temperature limits for heat-up, cooldown and hydrostatic test to account for the effects of irradiation of the reactor vessel materials for 6.6 effective full power years of operation.

Date of issuance: August 21, 1985.

Effective date: August 21, 1985.

Amendment No. 89.

Provisional Operating License No. DPR-20. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 3, 1985 (50 FR 27504).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Van Zoeren Library, Hope College, Holland, Michigan 49423.

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

Date of amendment request: February 15, 1985.

Brief description of amendment: The amendments revise the Technical Specifications to eliminate provisions that allow bypass of the high drywell pressure scram signal for the purpose of containment inerting and de-inerting.

Date of issuance: August 27, 1985.

Effective date: August 27, 1985.

Amendment No. 113 and 53.

Facility Operating Licenses Nos. DPR-57 and NPF-5. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 23, 1985 (50 FR 16004).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 27, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Applying County Public Library, 301 City Hall Drive, Baxley, Georgia.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50-366, Edwin I. Hatch Nuclear Plant, Unit No. 2, Appling County, Georgia

Date of amendment request: March 19, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to provide operating and surveillance requirements for automatic depressurization system bypass timers.

Date of issuance: August 27, 1985.

Effective date: August 27, 1985.

Amendment No. 52.

Facility Operating License No. NPF-5. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1985 (50 FR 20980).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 27, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: March 29, 1985.

Brief description of amendments: These amendments revise the Technical Specifications for the reactor trip system instrumentation and the engineered safety feature actuation system instrumentation. Of four groups of changes, the first provides criteria for when a channel needs to be adjusted following a heat balance, suspends the requirements for immediate shutdown when all trains of some instrumentation are inoperable, and changes the action statement when operable instrument channels are one less than the total number of available channels. The second group extends the period of time from one hour to two hours in which one channel of the reactor solid state protection system can be bypassed for surveillance testing. The last two groups are editorial in nature.

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment Nos.: 90 and 75.

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

Date of initial notice in **Federal Register:** May 21, 1985 (50 FR 20982).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: May 31, 1985, as supplemented June 7, 1985.

Brief description of amendments: The amendments revise the Technical Specifications by deleting the program and records retention requirements pertaining to environmental qualification of equipment.

Date of issuance: August 19, 1985.

Effective date: Within 30 days of date of issuance.

Amendment Nos.: 89 and 74.

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

Date of initial notice in **Federal Register:** July 17, 1985 (50 FR 29010).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 19, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 5, 1985, supplemented June 11, 1985 and modified June 20, 1985.

Brief description of amendment: The amendment modified the Maine Yankee Technical Specifications concerning Steam Generator Tube Surveillance Requirements.

Date of issuance: August 20, 1985.

Effective date: August 20, 1985.

Amendment No.: 84.

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

Date of initial notice in **Federal Register:** July 17, 1985 (50 FR 29006 at 29011).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 20, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, Wiscasset, Maine.

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

Date of application for amendment: June 6, 1985.

Brief description of amendment: The amendment added new technical specifications addressing the surveillance requirements related to the licensee's solid radioactive waste Process Control Program (PCP). Specifically, the requirements state that the PCP shall be used to verify the solidification of radioactive waste.

Date of issuance: August 22, 1985.

Effective date: October 3, 1985.

Amendment No.: 91

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 3, 1985 (50 FR 27502 at 27508)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: June 11, 1985.

Brief description of amendment: The amendment changed the testing frequency of the auxiliary feedwater pumps from quarterly to monthly.

Date of issuance: August 19, 1985.

Effective date: within 30 days of issuance.

Amendment No.: 90.

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 17, 1985 (50 FR 29006 at 29013).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 19, 1985.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: January 7, 1985, as amended April 1, 1985.

Brief description of amendment: The changes to the Technical Specifications permit reactor operation of Peach Bottom, Unit No. 3 with Reload No. 6 (Cycle 7).

Date of issuance: August 23, 1985.

Effective date: August 23, 1985.

Amendment No.: 114.

Facility Operating License No. DPR-56. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 27, 1985 (50 FR 7999) and May 21, 1985 (50 FR 20986).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 23, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania.

Power Authority of The State of New York, Docket No. 50-286, Indian Point Unit No. 3, Westchester County, New York

Date of application for amendment: March 27, 1985 and April 23, 1985, as supplemented August 1, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to allow the first of a three-phase fuel design transition from Westinghouse 15 × 15 low parasitic (LOPAR) design to the 15 × 15 Optimized Fuel Assembly (OFA) design with the introduction of Wet Annular Burnable Absorber (WABA) rods into the core and to allow an equivalent steam generator tube plugging level of up to 30% in any steam generator provided the equivalent average plugging level in all steam generators is less than or equal to 24%. The licensee's August 1, 1985 submittal provided additional information to the original amendment request and did not change the Technical Specifications.

Date of issuance: August 27, 1985.

Effective date: August 27, 1985.

Amendment No.: 61.

Facilities Operating License No. DPR-64. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1985 (50 FR 23549) and July 3, 1985 (50 FR 27508).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 27, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 9, 1985, as supplemented May 20 and June 20, 1985.

Brief description of amendment: The amendment modifies the Technical Specifications to delete the Boron Injection System.

Date of issuance: August 26, 1985.

Effective date: September 2, 1985.

Amendment No. 44.

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

Date of initial notice in **Federal Register**: July 17, 1985 (50 FR 29015).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, Garden and Washington Streets, Winnsboro, South Carolina 29180.

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

Date of application for amendment: September 17, 1984.

Brief description of amendment: The amendment adds to Technical Specifications 6.9.1.6 the requirement to report in the monthly operating report challenges to the pressurizer power operated relief value (PORV) and the pressurizer code safety valves.

Date of issuance: August 22, 1985.

Effective date: August 22, 1985.

Amendment No. 87.

Facility Operating License No. NPF-3. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 27, 1985 (50 FR 12165).

The Commission's related evaluation of the amendment is contained in a letter to the licensee dated August 22, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendment: October 8, 1984.

Brief description of amendment: The amendment adds Technical Specification Section 6.2.3. which requires administrative procedures to be developed and implemented to limit facility staff working hours.

Date of issuance: August 22, 1985.

Effective date: August 22, 1985.

Amendment No.: 88.

Facility Operating License No. NPF-3. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 27, 1985 (50 FR 12165).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 22, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendment: December 16, 1984.

Brief description of amendment: This amendment modifies paragraph 6.4.1 of the TSs to specify that the retraining and replacement training program for the facility staff is under the direction of the Nuclear Training Manager. Previously no position title was shown in paragraph 6.4.1.

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment No. 89

Facility Operating License No. NPF-3. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 27, 1985 (50 FR 12165)

The Commission's related evaluation of the amendment is contained in a letter to Toledo Edison Company dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606.

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

Date of application for amendments: February 9, 1984.

Brief description of amendments: The amendments add a description of the post-accident sampling program to the NA-1&2 TS Administrtive Controls, Section 6 in response to NUREG-0737 Item II.B.3 (Post-Accident Sampling) and II.F.1.2 (Sampling and Analysis of Plant Effluents).

Date of issuance: August 20, 1985.

Effective date: August 20, 1985.

Amendment Nos.: 65 and 50.

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical Specifications.

Date of Initial notice in **Federal Register**: June 4, 1985 (50 FR 23543 at 23554).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 20, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093, and Alderman Library, Manuscripts Department, Univesity of Virginia, Charlottesville, Virginia 22901.

Virginia Electric and Power Company, et al., Docket Nos. l50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: September 28, 1984.

Brief description of amendments: The amendments are administrative in nature and correct discrepancies presently existing in the NA-1&2 TS which relate to the Radiological Effluent Technical Specifications (RETS). The amendments add the Containment Vacumm Steam Ejector (Hogger) as a gaseous release path that is monitored and specify the figure for the Low Population Zone in the appropriate TS figure and correct numbers are assigned to appropriate TS Table numbers.

Date of issuance: August 29, 1985.

Effective date: August 29, 1985.

Amendments Nos.: 67 and 53.

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical specifications.

Date of initial notice **Federal Register:** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29. 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Board of Supervisors Office, Louisa County Courthouse, Louisa, Virgina 23093, and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

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

Date of application for amendments: April 15, 1985.

Brief description of amendments: The amendments revised the NA-1&2 TS 3/4 9.3 to specify a minimum decay time of 150 hours instead of the presently specified 100 hours prior to any movement of fuels for refueling operations.

Date of issuance: August 21, 1985.

Effective date: August 21, 1985.

Amendment Nos.: 66 and 52.

Facility Operating License Nos. NPF-4 and NPF-7. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 21, 1985 (50 FR 20969 at 20995).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 21, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093, and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia

Date of application for amendments: May 13, 1985.

Brief description of amendments: These amendments revise Technical Specifications Section 5.3 to modify the description of the fuel assemblies so that reconstituted assemblies may be placed into the core. In the reconstituted assemblies, fuel rods which are known to have failed have been removed and replaced with dummy (non-fueled) rods.

Date of issuance: August 26, 1985.

Effective date: August 26, 1985.

Amendment Nos. 102 and 102.

Facility Operating License Nos. DPR-32 and DPR-37. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 4, 1985 (50 FR 23555).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 26, 1985.

No significant hazards consideration comments received: No.

Local Public Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Virginia Electric and Power Company, et al., Docket No. 50-339, North Anna Power Station, Unit No. 2, Louisa County, Virginia

Date of application for amendment: February 11, 1985.

Brief description of amendment: The amendment provides relief from Surveillance Requirement 4.4.7 (Table 4.4-3) which requires that reactor coolant system chemistry limits for chlorides and fluorides be sampled on a continuing 72 hour basis. The relief from Surveillance Requirements 4.4.7 (Table 4.4-3) is applicable when the reactor coolant system is drained below the reactor pressure nozzle and the internals and/or head are in place. The relief is only applicable in Mode 6 (Refueling).

Date of issuance: August 21, 1985.

Effective date: August 21, 1985.

Amendment No.: 51.

Facility Operating License No. NPF-7: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 17, 1985 (50 FR 29006 at 29020).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Board of Supervisors Office, Louisa County Courthouse, Louisa, Virginia 23093, and the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

Dated at Bethesda, Maryland this 5th day of September 1985.

For the Nuclear Regulatory Commission.

Edward J. Butcher,

Acting Chief, Operating Reactors Branch #3, Division of Licensing.

[FR Doc. 85-21736 Filed 9-10-85; 8:45 am]

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