Docket No.: STN 50-454

OCT 3 1 1985

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: FEDERAL REGISTER NRC BI-WEEKLY NOTICES OF APPLICATIONS AND AMENDMENTS TO OPERATING LICENSES INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS BYRON STATION, 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 October 23, 1985.

Two Notices for the Byron Station, Unit 1 are contained in this issue of the publication. They are:

1. An amendment request which would revise the Technical Specifications to corret typographical and grammatical errors on six pages. This notice may be found on Page 43022, and

2. The Notice of Issuance of Amendment No. 1 to NPF-37 may be found on Page 43038.

Sincerely,

(S)

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

Enclosure:
Federal Register,
Dated October 23, 1985

cc: See next page

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Mr. Dennis L. Farrar
Commonwealth Edison Company

Byron Station
Units 1 and 2

cc:
Mr. William Kortier
Atomic Power Distribution
Westinghouse Electric Corporation
Post Office Box 355
Pittsburgh, Pennsylvania 15230

Michael Miller
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Dr. Bruce von Zellen
Department of Biological Sciences
Northern Illinois University
DeKalb, Illinois 61107

Mr. Edward R. Crass
Nuclear Safeguards & Licensing
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Mr. Julian Hinds
U. S. Nuclear Regulatory Commission
Byron/Resident Inspectors Offices
4448 German Church Road
Byron, Illinois 61010

Mr. Michael C. Parker, Chief
Division of Engineering
Illinois Department of
Nuclear Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Ms. Diane Chavez
528 Gregory Street
Rockford, Illinois 61108

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Joseph Gallo, Esq.
Isham, Lincoln & Beale
Suite 840
1120 Connecticut Avenue, N.W.
Washington, D. C. 20036

Douglass Cassel, Esq.
109 N. Dearborn Street
Suite 1300
Chicago, Illinois 60602

Ms. Pat Morrison
5568 Thunderidge Drive
Rockford, Illinois 61107

Ms. Lorraine Creek
Rt. 1, Box 182
Manteno, Illinois 60950

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to 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 October 9, 1985 (50 FR 41241), through October 11, 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 Rules and Procedures Branch, Division of Rules and Records, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Comments may also be delivered to Room 4000, Maryland National Bank Building, Bethesda, Maryland from 8:15 a.m. to 5:00 p.m., Monday through Friday.

By November 22, 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, N.W., 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 Atomic Safety and Licensing Board designated to rule on the petition and/or request, that the petitioner has made a substantial showing of good cause for the granting of a later petition and/or request. That determination will be based upon a balancing of the 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.

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

Date of amendment request: August 9, 1984, as supplemented August 9, 1985.

Description of amendment request: The amendment would change the Technical Specifications (TS) for the Standby Gas Treatment System (SBGTS) and the Control Room High Efficiency Air Filtration System (CRHEAF) as follows:

1. Obsolete footnotes granting relief from certain limiting conditions for operation (LCOs) during past periods of time would be deleted.

2. A requirement to verify analysis results within 31 days after carbon samples are removed would be added to the LCOs for both systems.

3. SBGTS surveillance requirements would be increased by adding detailed operating procedure (DOP) testing of high-efficiency particulate air (HEPA) filters and halogenated hydrocarbon testing of the charcoal adsorber banks every 18 months or following painting, fire or chemical releases that could contaminate the HEPA filters or charcoal adsorbers.

4. The requirement in TS 3.7.B.1.c for a daily demonstration that all active components of one SBGTS train are operable after the other train is found inoperable would be deleted, but such a demonstration within 2 hours would continue to be required.

5. Also in TS 3.7.B.1.c and in TS 3.7.B.1.e, the term "fuel handling" would be changed to "irradiated fuel handling, or new fuel handling over the spent fuel pool or core. . ." These changes are intended to clarify the intent of these LCOs, which limit reactor operation or fuel handling when SBGTS operation is impaired, and bring them into closer correspondence with Standard Technical Specifications.

6. The surveillance requirement in TS 4.7.B.1.a(2) to perform an instrument functional test on the humidistats controlling the SBGTS heaters would be deleted because these humidistats have been permanently bypassed. The humidistats were removed from service because they are not environmentally qualified and suitable replacements are not available.

7. An additional restriction "providing that within two hours all active components of the other CRHEAF train shall be demonstrated operable" would be added to Section 3.7.B.2.c, which permits reactor operation or refueling operations during the 7 succeeding days after one train of the CRHEAF is made or found incapable of supplying filtered air to the control room.

8. Section 4.7.B.2.c would be changed from requiring a demonstration of the operability of the CRHEAF heaters at rated power to a demonstration of the "ability of the heaters to perform their design function."

9. The words "once per 18 months" would be added to Section 4.7.B.3 to specify the time interval between instrument functional tests of the humidistat which controls the CRHEAF heaters.

10. The explanatory discussion in the BASES would be modified to reflect the above changes.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of its standards for determining whether license amendments involve significant hazards considerations by providing certain examples (48 FR 14870). One example of an amendment that is considered not likely to involve a significant hazard consideration is "(i) 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."

Proposed change no. 1 is similar to example (i) since it would eliminate footnotes that are no longer operative. Change no. 8 is similar to example (i) since its sole purpose is to restate the requirement so as to avoid possible misinterpretation that operability of the heaters is to be demonstrated with the reactor at rated power, whereas the intent is simply to determine that the heaters are functioning properly. Change no. 10 is also similar to example (i) since it involves only descriptive changes to achieve consistency.

Another Commission example of an amendment considered unlikely to involve a significant hazard consideration is "(ii) a change that constitutes an additional limitation, restriction or control not presently included in the technical specifications: for example, a more stringent surveillance requirement." Proposed change nos. 2, 3, 5, 7 and 9 would impose such additional requirements and are, therefore, similar to example (ii).

Another example of an amendment considered unlikely to involve a significant hazards consideration is "(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 acceptance criteria with respect to the system or component specific 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."

Proposed change no. 4 would eliminate daily testing which is no longer considered necessary to verify that the remaining SBGTS train is operable during the 7-day period the plant is allowed to continue operation while the inoperable train is being repaired. Reducing the number of such tests will reduce the resulting wear on the SBGTS components and thereby provide greater assurance that they will operate properly. While the proposed change would relax the existing surveillance requirements, it meets all acceptance criteria of the Standard Review Plan and, therefore, is similar to example (vi) above.

Proposed change no. 6 does not compromise safety because the humidistats are not essential. The licensee states that the relative humidity of the incoming gas stream to the SBGTS will continue to be controlled by the heaters, which are now being energized when the exhaust fans are energized. Without the humidistats, the heaters will be in operation more of the time and wear out faster, which may in some way reduce a safety margin but where the results would be within acceptance criteria of the Standard Review Plan. This change is, therefore, similar to example (vi).

Having found that all of the changes included in this proposed amendment are similar to examples considered not likely to involve significant hazards considerations, the staff has made a proposed determination that the amendment request involves no significant hazards consideration.

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

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

NRC Branch Chief: Domenic B. Vassallo.

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station Units 1 and 2, Ogle County,

Illinois Date of application for amendment: September 27, 1985.

Description of amendment request: The amendment would revise the Technical Specifications to correct typographical and grammatical errors on six pages.

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 of actions not likely to involve a significant hazards consideration (48 FR 14870). One of the examples (i) relates to purely administrative changes to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correct errors, or change nomenclature. The proposed change would correct typographical and grammatical errors. Based on the above, since the proposed change involves actions that conform to example (i), the staff proposes to determine that this application for amendment involves no significant hazards consideration.

Local Public Document Room locations: Rockford Public Library, 215 N. Wyman Street, Rockford, Illinois 61103.

Attorney for licensee: Michael Miller, Isham, Lincoln & Beale, One First National Plaza, 42nd Floor, Chicago, Illinois 60603.

NRC Branch Chief: B. J. Youngblood.

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

Date of amendment request: August 13, 1985.

Description of amendment request: The proposed amendments would delete license conditions 3.N.1, 3.N.2 and 3.N.3 from Provisional Operating License No. DPR-19 for Dresden Unit 2 and 3.M.1, 3.M.2 and 3.M.3 from Facility Operating License No. DPR-25 for Dresden Unit 3 and transfer the requirements therein to appropriate sections of the respective Technical Specifications for the units. The transfer of requirements would be either the same technically or in an equivalent or improved amended form. The aforementioned license conditions all involve the spent fuel storage racks and the spent fuel pool. License condition 3.M.4 of Facility Operating License DPR-25 is proposed to be deleted entirely as it requires conditions to be reflected in the Dresden Updated FSAR which have now been included in the latter document.

Basis for proposed no significant hazards consideration determination: The standards used to arrive at a proposed determination that a request for amendments involves no significant hazards consideration are included in the Commission's regulations, 10 CFR 50.92, which state that the operation of the facilities in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee, in the August 13, 1985 submittal, addressed these criteria as follows:

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

(1) The transfer of License Conditions 3.N and 3.M for Units 2 and 3 respectively into Appendix A is an administrative change which does not in any way change the licensing requirements, operating practices or equipment reliability at the facility.

(2) The addition of an allowed capacity for the new-fuel storage vault only reflects the existing storage capacity and does not increase the number of bundles stored over that previously allowed.

(3) The use of K-INF criteria in place of the U-235 axial loading criteria is an alternate means of specifying reactivity limits for fuel bundles in storage and does not change the manner in which fuel is handled or stored. Reactivity restrictions are provided to protect against fuel pool criticality; this protection is maintained by the proposed K-INF limits.

The proposed amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated because all three of the proposed changes are largely administrative and deal with the manner in which compliance with fuel storage requirements will be demonstrated. The proposed changes do not allow any new or different modes of operation nor any changes to plant equipment.

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

(1) The transfer of License Conditions into the Technical Specifications is administrative and does not affect the manner in which the plant will be operated.

(2) The addition of an allowable capacity for the new-fuel vault represents an additional restriction not previously included in the Technical Specifications. The allowable number of bundles does not reflect a change or increase in the storage capacity of the plant.

(3) The substitution of bundle K-INF limits for the U-235 axial loading restriction reflects a more sophisticated method for identifying bundle reactivities. Compliance with these limits will assure that future fuel designs stored in the spent fuel pool are bounded by the pool criticality analyses which have been performed. These analyses have demonstrated that the

margin of safety for pool criticality, i.e., pool K_{eff} less than or equal to 0.95, is maintained.

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

Local Public Document Room location: Morris Public Library, 604 Liberty Street, Morris, Illinois 60451.

Attorney for licensee: Robert G. Fitzgibbons, Jr., Isham, Lincoln and Beale, Three First National Plaza, Suite 5200, Chicago, Illinois 60602.

NRC Branch Chief: John A. Zwolinski.

Commonwealth Edison Company, Docket No. 50-373, La Salle County Station, Unit 1, La Salle County, Illinois

Date of amendment request: October 2, 1985.

Description of amendment request: The proposed amendment to operating License NPF-11 would revise the La Salle Unit 1 Technical Specifications because the eight 26-in and two 8-inch vent and purge isolation valves are being replaced by Clow Corporation made valves which meet all the requirements for containment vent and purge isolation valves. Since the new valves are qualified to close from any position including the full open (90°) position Technical Specifications 3.6.1.8, 4.6.1.8 and associated basis 3/4.6.1.8 must be revised to remove the 50° limit on valve opening. This limit was required until these valves could be replaced by valves capable of closing during a loss-of-coolant accident or a steam line break. In addition, the new valves do not contain resilient seals. As a result, the once per 92 days leakage surveillance is no longer required since the purpose of the accelerated leakage rate testing (every 92 days) was to provide an early indication of resilient material seal degradation.

The above items addressed in this proposed amendment and these modifications will be incorporated at the first refueling outage in accordance with Supplement No. 7 to the La Salle Safety Evaluation Report. *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 has determined and the NRC staff agrees that the proposed amendments will not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the new vent and purge isolation valves replace the existing isolation valves one for one. No additional valves have been added. The new valves meet the requirements for vent and purge containment isolation valves. This amendment simply removes requirements which only apply to the valves being removed.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because the modification does not affect the containment isolation valve arrangement.

(3) Involve a significant reduction in the margin of safety because the design continues to meet the requirements of General Design Criterion 56, as specified in the updated Final Safety Analysis Report. Accordingly, the Commission proposes to determine that the proposed changes to the Technical Specifications involve no significant hazards considerations.

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Ogelsby, Illinois 61348.

Attorney for licensee: Isham, Lincoln and Burke, Suite 840, 1120 Connecticut Avenue, N.W., Washington, DC 20036.

NRC Branch Chief: W. R. Butler.

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 16, 1985, as supplemented September 20 and 23, 1985.

Description of amendment request: The proposed amendments would change Technical Specification 3.7.7 and its associated bases to reflect that the Auxiliary Building Filtered Ventilation Exhaust (VA) system consists of two shared safety-grade systems serving the common Auxiliary Building, rather than one safety-grade system for each of the two McGuire units. The proposed change would require, as a limiting condition for operation, that both VA systems be operable when either McGuire Unit 1 or Unit 2 is in Modes 1 (power operation), 2 (startup), 3 (hot standby), or 4 (hot shutdown). A change in the action statement would increase the time allowed to restore the VA system to operable status (when one of the two systems is inoperable) from 24 hours to 7 days, and the action statement would be clarified to reflect its applicability to both Unit 1 and Unit 2. Consequently, if one of the two systems should not be restored to operable status within the allowed 7 days, both McGuire Unit 1 and Unit 2 would be required to be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours. The proposed change would also delete an outdated footnote for Specification 3.7.7 which allowed hot standby conditions to be maintained until 11:59 p.m., September 7, 1983.

Basis for proposed no significant hazards consideration determination: The function of the VA system is to filter radioactive materials associated with coolant leakage from ECCS equipment in the Auxiliary Building (shared for both units) following a LOCA. The VA system includes for 50% capacity fans (two associated with Unit 1 and two with Unit 2); two filter trains (one associated with each unit); and two trains of ductwork (one associated with each unit). Air intakes for these two VA trains are located in the same general open area of the Auxiliary Building near the ECCS Pump Rooms. Both VA trains are automatically actuated following a LOCA in either unit and are powered from separate sources.

The Commission's Standard Technical Specifications (STS) for Westinghouse plants (NUREG-0452, Revision 4, Specification 3.7.8) provide a 7 day restoration period for filtration designs with redundant systems if one of the two redundant systems remains functional. The licensee has determined that the McGuire VA system design meets the requirements of a redundant system for the common Auxiliary Building area, that the consequences of inoperability of one of the two VA system trains following a LOCA are insignificant, and that the McGuire Specifications should be revised in accordance with this STS. The licensee also notes that the offsite thyroid dose calculated for a LOCA and presented in FSAR Section 15.6.4.3 and Table 15.6.4-11 (i.e., 200 REM at the exclusion area boundary which includes the contribution due to ECCS equipment leakages) took no credit for exhaust filtration by the VA system, and still these consequences were well below 10 CFR Part 100 values. The licensee has also determined that the probability of a LOCA with fuel damage during a 7-day period is less than 10^{-6} and the probability of such occurrence in combination with significant ECCS equipment leakage is even smaller. Preliminary review results and separate calculations by the Commission support these statements and conclusions by the licensee.

The Commission has provided certain examples (48 FR 14870) of actions likely to involve no significant hazards considerations. The licensee's request to allow 7 days rather than 24 hours to restore one inoperable VA system does not match any of those examples. Based on the review of the licensee's submittal, the staff proposes to determine that this part of the licensee's amendment request does not involve a significant hazards consideration. Operation with this requested change would not involve a significant increase in the probability of an accident previously analyzed or create the possibility of a new or different kind of accident from any accident previously analyzed because the system serves only to mitigate accidents, and because the duration of the allowed time (7 days) is sufficiently limited such that the attendant opportunity for a LOCA is so small as to be negligible. This change from 24 hours to 7 days does not involve a significant reduction in a margin of safety or a significant increase in the consequences of an accident previously evaluated because as discussed above either one of the two redundant filtration trains would accomplish the system function, and even if both filter trains should fail to function, the increment is such that the total offsite doses to the thyroid following a LOCA would remain well below the 10 CFR Part 100 guideline values.

One of the Commission's examples of an amendment likely to involve no significant hazards consideration relates to changes (ii) that constitute additional limitations, restrictions, or controls not presently in the Technical Specifications. The changes to clarify that the action statement applies to both units is a more appropriate representation of system (shared) design and provides a more restrictive requirement (dual unit shutdowns) if one inoperable VA system is not restored within the allowed time period. Another example (i) involves purely administrative changes to Technical Specifications. The change to delete the existing outdated footnote is purely administrative and has no safety implication.

On the above bases, the Commission proposes to determine that these 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, Dockets Nos. 50-269, 50-270, and 50-287, Oconee Nuclear Station, Units Nos. 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: July 26, 1985.

Description of amendment request: The proposed amendments would revise the Station's common Technical Specifications to add Limiting Conditions for Operation (LCO), surveillance requirements and bases, and manpower requirements for the operation of the Standby Shutdown Facility (SSF). The SSF is an alternate means to provide the capability to maintain each Oconee unit as hot shutdown. It is a facility which would mitigate the effects of postulated fires within certain fire areas. In addition, the facility provides the means in which the safe shutdown requirements of turbine building flooding and physical security are resolved.

Specification 3.18 provides the LCO for the SSF. The systems of the SSF necessary to assure its operability, namely the SSF Auxiliary Service Water (ASW), SSF Reactor Coolant (RC) Makeup, associated instrumentation, electrical generation and distribution are included. Specification 3.18.1 requires that these systems be operable for each Unit in the hot shutdown, hot standby, or power operation. Specification 3.18.2 addresses the SSF ASW system, covering planned test or maintenance, restoration to operable status if inoperable. In a similar manner, Specifications 3.18.3, 3.18.4, and 3.18.5 cover the SSF RC Makeup, the SSF Power System and associated SSF instrumentation, respectively.

Specification 4.20 provides the surveillance requirements for the SSF. Pumps, valves, instrumentation, and electrical power systems are included. The pumps and valves required for the SSF systems to function are included in the pump and valve test program which is maintained in accordance with ASME Section XI. SSF instrumentation is both checked and calibrated on frequencies contained in Table 4.20-1. The periodic surveillance is frequent enough to provide assurance that the instrumentation is properly functioning. Calibrations are conducted on either an annual or refueling outage interval depending on location of the device and whether or not it's accessible during operation. Specification 4.20.3 covers operability of the SSF diesel generator and SSF DC power system.

Finally, Specification 6.1 is revised to include the manpower requirements for the operation of the SSF.

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). Example (ii) of the types of amendments not likely to involve significant hazards considerations applies in this case as these amendments constitute an additional limitation, restriction or control not presently included in the Technical Specifications.

The Oconee SSF was designed to resolve the safe shutdown requirements for fire protection, turbine building flooding, and physical security requirements. The NRC has reviewed the design and provided the results of this review in a letter dated April 28, 1983. These proposed license amendments are being submitted by the licensee in response to an NRC request contained in the April 28, 1983 letter.

The current Technical Specifications do not include operability nor surveillance requirements for the SSF. Therefore, the proposed amendments match the example.

Accordingly, the Commission proposes to determine that the amendment changes do not involve significant hazards considerations.

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

Attorney for licensee: J. Michael McGarry, III; Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, N.W., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: July 29, 1985.

Description of amendment request: The proposed amendments would revise the Station's common Technical Specifications (TSs) to delete TS 4.2.4, 4.2.5 and Table 4.2-1 on the Reactor Vessel Material Surveillance program. By letter dated May 8, 1985, the NRC had informed the license that the Babcock and Wilcox Owners Group (B&WOG) Materials Committee Report, BAW-1543, Revision 2 and 2A, "Integrated Materials Vessel Surveillance Program, February 1984." would be acceptable for referencing in

Oconee Nuclear Station license applications in accordance with the requirements of Section II.C of Appendix H. 10 CFR 50.

Currently, Oconee 1, 2 and 3 have Technical Specification requirements for reactor vessel materials surveillance which satisfy Appendix H, 10 CFR 50, and which are a part of the B&WOG Materials Committee integrated reactor vessel materials surveillance program. As a result of the NRC acceptance of BAW-1543, Revision 2 and 2A, to satisfy the requirements of Appendix H, 10 CFR 50, it is not considered necessary to maintain the current reactor vessel material surveillance requirements within the Oconee Technical Specifications. Therefore, in accordance with Section II.C of Appendix H, 10 CFR 50, the licensee has submitted for NRC consideration and approval, the B&WOG integrated reactor vessel materials program, BAW-1543 Revision 2 and 2A, for Oconee Units 1, 2 and 3. This document will be maintained current and will serve as the basis for reactor vessel materials surveillance program for Oconee Nuclear Station. Subsequent changes and/or revisions to the program will be made through revision of BAW-1543. The licensee will notify the NRC staff of such changes and will request approval for use of the modified integrated surveillance program.

Basis for proposed no significant hazards consideration determination: The NRC staff has made a proposed determination that these amendment requests involve no significant hazards consideration by applying the standards established by the Commission's regulations in 10 CFR 50.92. This ensures 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 new or different kind of accident from any accident previously evaluated; or

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

The proposed Technical Specification amendments reflect the new process in which changes to Oconee's Reactor Vessel Surveillance Program (RVSP) will be handled in the future. The current Oconee Nuclear Station Technical Specifications 4.2.4 and 4.2.5 for reactor vessel materials surveillance satisfy the requirements of Appendix H, 10 CFR 50. However, as part of the Babcock and Wilcox Owners Group (B&WOG) Materials Committee integrated reactor vessel materials surveillance program, these Technical Specificatons are affected by changes in the program.

By a letter dated May 8, 1985, NRC found the B&WOG Materials Committee Report, BAW-1543, Revision 2 and 2A, "Intergrated Materials Vessel Surveillance Program, February 1984," acceptable for referencing in Oconee Nuclear Station license applications in accordance with Section II.C of Appendix H, 10 CFR 50. This document provides the basis for and explains the Oconee Nuclear Station reactor vessel materials surveillance program incuding the Surveillance Capsule Insertion and Withdrawal schedule. This document will be maintained current to reflect changes in the program. Subsequent changes or revisions to the program will be made through revision of BAW-1543. If affected by the change, the licensee will request the NRC approval for use of the modified integrated surveillance program for Oconee Nuclear Station per Section II.C of Appendix H of 10 CFR 50.

Inasmuch as the proposed Technical Specification change is in support of this program and that the NRC staff has accepted the BAW-1543 and found it applicable for Oconee Nuclear Station reactor vessel surveillance program, it is considered unnecessary to retain Technical Specifications 4.2.4, 4.2.5 and Table 4.2-1.

The NRC staff has determined, based on the consideration that the requested amendments will not alter the Oconee reactor vessel surveillance program, which is in compliance with the regulations, that the revisions do not involve a signifiant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident and will not involve a significant decrease in a safety margin. Therefore, the Commission proposes to determine that the changes do not involve significant hazards considerations.

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

Attorney for licensee: J. Michael McGarry, III, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, N.W., Washnington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: August 15, 1985.

Description of amendment request: The proposed amendments would revise the Station's common Technical Specifications (TSs) to correct typographical errors in several sections; correct a section title in the Table of Contents; address a change in nomenclature; update Final Safety Analysis Report (FSAR) references; delete out-of-date footnotes; delete an unnecessary section; change wording for clarification; and also, update organizational charts that appear in the Technical Specifications.

There are several areas where typographical errors were found in the Oconee Technical Specifications. High pressure valves are designated as HP, but were mistakenly referred to as 3HP in two places. The word "and" was used instead of "or" in section 3.7.1. and thirdly, an "f" is shown instead of a "g" in a reference in section 3.8.2(e). In section 3, the word "present" was misspelled, and an underline was used instead of a minus sign to denote "±". Finally, in section 3.5.2., the words that relate to the acronym APSR were incorrect and are now being corrected. The changes included in the proposed amendments correct these errors.

An inconsistency was found between the Table of Contents and the title for section 1.2.3. The Table of Contents refers to "Reactor Control" when it should be "Reactor Critical". This change will provide for uniformity throughout the Technical Specifications, and thus, assure a consistent application of the term.

The initial Oconee FSAR update was provided as required by 10 CFR 50.71 by the licensee's letter date July 19, 1982. The updated FSAR was reformatted to be consistent with present FSAR format criteria. This resulted in the FSAR references within the Technical Specifications being out of date. The updating of the reference to the FSAR within the Technical Specifications assures that the appropriate figure of the FSAR is being identified. The updating of the Technical Specifications is an administrative change to achieve consistency with other documents.

In section 6.1.1.4 of the Technical Specifications, a change in nomenclature is requested. The Health Physicists at the Oconee Nuclear Station are referred to as Station Health Physicists, not Site Health Physicists.

Two footnoted special exemptions should be deleted as they are no longer applicable. In both cases, the dates of which the footnotes are valid have passed; therefore they can be deleted.

In sections 6.1.3 and 6.6.2, some wording has been changed in order to achieve clarity and consistency throughout the Technical Specifications. "Individuals" was changed to "members of the public" to clarify which individuals and "during the reporting period" is being used instead of "each quarter" and "each calender quarter" to be consistent with other Techical Specifications. "Container volume" was changed to "total container volume, in cubic meters", for clarification purposes. Also, since 10 CFR 61 curently does not address types of containers, "type of container" was changed to "numbers of shipments". Finally, a footnote was added concerning Radioactive Effluent Release Reports to achieve consistency with other Technical Specifications.

Technical Specification 3.1.8, Single Loop Restriction, is being deleted because it is obsolete and no longer applicable to Oconee. There are currently no plans to ever use this specification at Oconee. The original purpose for this section was to (1) supplement the 1/6 scale model test information, (2) verify predicted flow through the idle loop, (3) verify that changes in power level did not affect flow distribution or core power distribution and (4) demonstrate that the limiting safety system settings (pump monitor trip setpoint and reactor outlet temperature trip setpoint) could be conservatively adjusted taking into account instrument errors. In addition, this specification required prior Commission approval before it could be used.

In summary, this specification was included in Oconee Technical Specification to provide additional restrictions for single loop operation solely for the purpose of performing tests. During routine operations, single loop operation restriction is provided by Specification 2.3 Specification 3.1.8 is limited to when special tests are performed, and in addition required prior Commission approval. Thus, the deletion of Specification 3.1.8 would not result in the removal or decrease in any limitation, restriction or control. In addition, the reference to single loop restrictions in section 6.6.3 is being deleted.

The final revisions are updates to the Station Organizational Chart and Management Organization Chart for Oconee Nuclear Station to achieve consistency with Duke Power's current organization.

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). Example (i) of the types of amendments not likely to involve significant hazards considerations is an amendment considered to be purely administrative. For example, a change to achieve consistency throughout the technical specification, correction of an error, or a change in nomenclature.

The proposed Technical Specification changes have been determined by the Commission to contain administrative changes only. The requested changes are required so that the Technical Specifications will be consistent throughout and past omissions will be corrected.

Briefly, the proposed amendments correct typographical errors in several places; revise the Table of Contents to provide for consistency; change nomenclature in one place; update two organizational charts; update FSAR figures and tables being referenced; delete out-of-date footnotes; and change wording for clarification.

The reason for the deletion of Technical Specification 3.1.8. Single Loop Restriction, is because it is obsolete and no longer applicable to Oconee. Further, there are currently no plans to ever use this specification at Ocone. This specification was included to provide, during special tests being conducted, additional restrictions for single loop operation. During routine operations, single loop operation restriction is provided by Specification 2.3. In addition, prior to invoking Specification 3.1.8, specific Commission approval was required. Thus, the deletion of Specification 3.1.8 would not result in the removal or reduction in any limitation, restriction, control or margin of safety.

The Commission has determined, based on the above consideration that the requested amendments are administrative in nature that the proposed license amendments appear to be encompassed by example (i) of amendments not likely to involve significant hazards consideration. On this basis, the Commission proposes to determine that these amendments do not involve significant hazards considerations.

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

Attorney for licenses: J. Michael McGarry, III, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street NW., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: August 22, 1985.

Description of amendment request: The proposed amendments would revise the Station's common Technical Specification (TSs) to correct a typographical error, delete an expired footnote, update the station organization by adding the Station Services and Integrated Scheduling areas, and provide clarity and consistency through different wording.

A footnote is being deleted from Section 3.3.5.c(1)(b). The footnote was no longer valid after April 20, 1985. There is also a typographical error in this section that is being corrected.

Technical Specifications 6.1.2.1.h. and i. require *annual* review of the station security program, the station emergency plans and their implementing procedues. The wording is being changed to read "once per 12 months" instead of "annually."

Technical Specifications 6.1.2.1.h. also required that the Station Manager approve all procedure changes in the security program implementing procedures. This specification is being changed to allow the Station Services Superintendent to also approve changes.

The Superintendent of Integrated Scheduling and the Station Services Superintendent are included in Specifications 6.1.1.3 and 6.1.2.1.a, c. and e. and 6.2.2. The proposed changes would allow the Station Services Superintendent and the Superintendent of Integrated Scheduling to review and/or approve procedures specified under Specification 6.4 and changes thereto (6.1.2.1.a.), modifications of safety-rated structures, systems or components (6.1.2.1.c.), proposed tests and experiments which affect nuclear safety and are not addressed in the Final Safety Analysis Report or Technical Specifications (6.1.2.1.e.), and Reportable Events (6.2.2), if so designated by the Station Manager.

Also, in section 6.2.1, the wording is being changed to better reflect the Station Manager's role in the occurrence of a reportable event. the Station Manager does not investigate a reportable event himself, but instead sees that the event is investigated by the appropriate personnel.

Section 6.2.2 is being revised to include the Superintendent of Integrated Scheduling and Station Services Superintendent.

Finally, Section 6.2.3 is being revised for completeness. Reportable events are reported pursuant to Specification 6.6.2 and 10 CFR 50.73.

Basic 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). Example (i) of the types of amendments not likely to involve significant hazards considerations in an amendment considered to be a purely administrative change to the Technical Specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

The proposed Technical Specifications have been determined to contain administrative changes only. The requested changes are required so that the Technical Specifications will be consistent throughout and consistent with the Administrative Policy Manual for Nuclear Stations.

The Commission has determined, based on the above consideration, that the requested amendments are administrative in nature. Thus, the proposed license amendments appear to be encompassed by example (i) of amendments not likely to involve significant hazards consideration. On this basis, the Commission proposes to determine that these amendments do not involve significant hazards considerations.

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

Attorney for licensee: J. Michael McGarry, III, Bishop, Liberman, Cook, Purcell and Reynolds, 1200 17th Street, N.W., Washington, D.C. 20036.

NRC Branch Chief: John F. Stolz.

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

Date of amendment request: August 23, 1985, revising the October 22, 1984, submittal.

Description of amendment request: The proposed amendment requests approval for changes to the Appendix A Technical Specifications (TS) related to the Reactor Coolant System Leakage in Sections 1., 3.3 and 4.3 of the TS by (1) the addition of reactor coolant leak rate detection requirements and surveillance, (2) the incorporation of additional requirements for identified and unidentified leakage, (3) the addition of definitions for identified and unidentified leakage, and (4) the correction of the Bases to Section 3.3, Reactor Coolant, to reflect the actual plant configuration.

Basis for proposed no significant hazards consideration determination: The licensee, in its submittal dated October 22, 1984, proposed additional TS on reactor coolant leakage to incorporate the requirements of Section 4.16.2 in the Integrated Plant Safety Assessment Report, NUREG–0822 dated January 1983, for Oyster Creek and of IE Bulletin 82–03. Section 4.16.2 stated that the TS do not contain requirements regarding the leakage detection systems and that the licensee committed to more restrictive TS requirements for unidentified leakage in its final response to IE Bulletin 82–03. The licensee's October 22, 1984, submittal addresses Section 4.16.2 and IE Bulletin 82–03 by requesting additional requirements in Sections 3.3 and 4.3, Reactor Coolant, of the TS on the following: leakage from the reactor coolant system and the reactor coolant leakage detection systems. It also requests corrections to the Bases for Section 3.3 to have the Bases reflect the actual plant configuration.

The October 22, 1984, request was noticed in the **Federal Register** on February 27, 1985 (50 FR 7990) as an additional limitation, restriction, or control not presently included in the TS and is, therefore, consistent with example (ii) of the Commission's guidance (48 FR 14870, April 6, 1983) as a type of action which would not likely involve a significant hazards consideration and the staff proposed to determine that the requested action would not involve a significant hazards consideration.

The August 23, 1985, submittal has all the TS proposed in the October 22, 1984, submittal and additionally revises the proposed TS 3.3.D.1.c and 3.3.D.3 on the rate of increase of unidentified leakage. This revision was the result of discussions between the licensee and the staff and was to bring the proposed TS into agreement with the Standard Technical Specifications for General Electric Boiling Water Reactors, NUREG–0123, Revision 3. The revised TS 3.3.D.1.c and 3.3.D.3 remain additional restrictions on plant operation not presently included in the TS.

The proposed TS in the August 23, 1985, submittal, therefore, would also constitute an additional limitation, restriction, or control not presently included in the TS and is, therefore, consistent with example (ii) of the Commission's guidance as a type of action which would not likely involve a significant hazards consideration. Therefore, the staff proposes to determine that the requested action would not involve a significant hazards consideration.

Local Public Document Room location: Ocean County Library, 101 Washington Street, Toms River, New Jersey 08753.

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

NRC Branch Chief: John A. Zwolinski.

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

Date of amendment request: September 30, 1985.

Description of amendment request: Requests approval of changes to the Appendix A Technical Specifications (TS) to revise Table 4.1.1, Minimum Check, Calibration and Test Frequency for Protective Instrumentation, in Section 4.1, Protective Instrumentation, of the TS. The changes would delete the requirement for channel check for the following instrument channels: low reactor water level and low-low reactor water level, due to a replacement of instruments in accordance with 10 CFR 50.49(g).

Basis for proposed no significant hazards consideration determination: The TS Table 4.1.1 requires that a daily channel check be performed on the low reactor water level and low-low reactor water level instrument channels. Channel check is defined in the TS (Definition 1.19A) as a qualitative determination of acceptable operability by observation of channel behavior during operation. Switches in these two channels are currently equipped with indicating gauges; however, during the Cycle 10M outage scheduled to begin in October 1985, these non-environmentally qualified switches are to be replaced with qualified switches. The qualified switches are not equipped with indicating gauges. Therefore, a channel check cannot be made on these channels after the new switches are installed.

The non-environmentally qualified switches are being replaced by qualified switches to meet the schedule and technical requirements of 10 CFR 50.49(g) and the staff's letter of March 30, 1985, to have all electrical equipment at Oyster Creek important to safety environmentally qualified by November 30, 1985.

The new switches will perform the same safety function as the switches they replace. These new switches are similar to switches in other instrument channels listed in Table 4.1.1 which do not allow a channel check of the instrument channel. These other channels have an "NA" (not applicable) listed under the column for channel check in Table 4.1.1.

The daily channel check does not verify the channel's proper response or that it responds within acceptable range and accuracy to fulfill its safety functions. The channel check is only the qualitative determination of acceptable operability of the channel by comparing, in this case, the existing channel switches indicating gauges to each other. Tests of proper functioning of an instrument channel are performed by the channel calibration and channel test which are also listed in Table 4.1.1. The frequency for channel calibration and channel test would not be changed by the licensee's proposed action.

An instrument channel for which a channel check cannot be performed is within acceptable criteria with respect to the reactor protection system as specified in both the Standard Review Plan, Section 7.2, Reactor Trip System, and in the Integrated Plant Safety Assessment Report (NUREG-0822 dated January 1983) for Oyster Creek for the staff's Systematic Evaluation Program. In addition, similar instrument channels to the low reactor water level and low-low reactor water level in the reactor protection system lack the capability of a channel check. Although the channel check or lack of it does not affect the probability of a previously analyzed accident and does not introduce an accident not previously analyzed, it may increase the consequences of a previously analyzed accident or may reduce a safety margin because a qualitative determination of acceptable operability by observation of channel behavior may indicate the channels are not functioning properly. However, there are other instrument channels of the reactor protection system available to respond to an accident to provide a defense-in-depth. Therefore, this proposed change is a change which may result in some increase to the consequences of a previously analyzed 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. Thus, this proposed change is encompassed by the Commission's example (vi) provided in 48 FR 14870 of actions not likely to involve significant hazards considerations. Based on this, the staff proposes to determine that the requested action involves no significant hazards consideration.

Local Public Document Room location: Ocean County Library, 101 Washington Street, Toms River, New Jersey 08753.

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

NRC Branch Chief: John A. Zwolinski.

Louisiana Power and Light Company, Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of Amendment Request: August 1, 1985.

Description of Amendment Request: The proposed change would revise the Appendix A Technical Specifications by correcting three typographical errors in Table 3.8-1, "Containment Penetration Conductor Overcurrent Protective Devices" of Technical Specification 3/4.84, "Electrical Protective Devices".

Technical Specification 3/4.8.4 delineates the operability and surveillance requirements for the containment penetration conductor overcurrent protective devices listed in Table 3.8-1. The requirements of this Technical Specification ensure these devices will not prevent safety related valves from performing their function. The proposed change to Table 3.8-1 consists of the following three parts:

(a) Item 8, 480 Volts Power from MCCs, Table 3.8-1, page 3/4 8-24 currently lists the valve number as 1SI-V1508 TK 1B. The proposed change will correct the typographical error in the tank designation suffix to accurately list the valve number as 1SI-V1508 TK 2B.

(b) Item 57, 120 Volts Control Power from PDPs or MCCs, Table 3.8-1, page 3/4 8-39 currently lists the Power Distribution and Motor Data (PDMD) sheet number for primary protection as 148. The proposed change will correct the typographical error to accurately list the PDMD sheet number as 148A.

(c) Item 71, 120 Volts Control Power from PDPs or MCCs, Table 3.8-1 page 3/4 8-41 currently lists the valve number as 2BM-P237. The proposed change will correct the typographical error in the code class prefix to accurately list the valve number as 7BM-P237.

Basis for Proposed No Significant Hazards Considerations Determination: The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (49 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to a purely administrative change to technical specifications, correction of an error, or change in nomenclature.

The proposed changes to Table 3.8-1 as described in parts a, b, and c above, will correct the typographical errors and bring the Technical Specification into conformance with other plant documents. Therefore, the proposed changes are similar to example (i).

As the changes requested by the licensee's August 1, 1985 submittal fit the example provided, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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

Attorney for licensee: Mr. Bruce W. Churchill, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M St., NW., Washington, D.C. 20036.

NRC Branch Chief: George W. Knighton.

Louisiana Power and Light Company, Docket No. 509-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of Amendment Request: August 1, 1985.

Description of Amendment Request: The proposed change would revise the Appendix A Technical Specifications by changing the first inservice inspection period for inaccessible snubbers in Technical Specification 3/4.7.8 "Snubbers".

Technical Specification 4.7.8 delineates the surveillance requirements for hydraulic and mechanical snubbers. In particular, item (b) allows for independent inspection of accessible and inaccessible snubbers, and requires that the first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing "POWER OPERATION" and shall include all hydraulic and mechanical snubbers.

Waterford 3 power operation commenced on March 18, 1985 placing the beginning of the initial snubber inservice visual inspection period at July 18, 1985. However, in order to take advantage of an unscheduled outage, Louisiana Power and Light Company (LP&L) performed an inservice visual inspection of inaccessible hydraulic and mechanical snubbers during mid-June, 1985—approximately 3 months after commencing power operation.

The requested Technical Specification change would alter the beginning of the first inservice visual inspection period from four months to two months post-power operation for inaccessible snubbers only. Technical Specification 4.7.8.b would be footnoted to reflect the change. With this change LP&L will be allowed to take credit for the June 1985 visual inspection of inaccessible snubbers, precluding a potential future plant shutdown during the 4-10 month period that may have been required for inaccessible snubber inspection.

Basis for Proposed No Significant Hazards Considerations Determination: The Commission has provided guidance concerning the application of 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 changes are clearly with all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

The proposed change allows LP&L to take credit for a visual inspection of inaccessible snubbers conducted approximately three months after commencing power operation rather than the four months required by the existing Technical Specification. The time period from initial power operation to the beginning of the visual inspection period is intended to ensure exposure of the snubbers to representative plant conditions.

The operating history of Waterford 3 over the initial three-month period covers several heat-ups and cool-downs along with numerous plant trips, both planned and inadvertent. This three-month history constitutes a representative exposure to plant conditions for validation of the initial snubber inspection and validation of snubber operability. An additional month's delay of the initial inspection to mid-July provides little additional exposure (one heat-up and several inadvertent trips) due to outages experienced during that time.

Additionally, the proposed change is in conformance with ANSI/ASME Standard OM4–1982, "Dynamic Restraints Examination and Performance Testing". Section 3.2.3, Inservice Examination Frequency, states: "The initial inservice examination of all snubbers shall be initiated after at least 2 months of power operation and shall be completed prior to 12 calendar months after initial criticality."

Based on the low system demands occurring during the fourth month of power operation, and the technical guidance of ANSI/ASME OM4–1982, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change deals only with a scheduling interval and introduces no new systems, procedures or modes of operation. As discussed above, the inaccessible snubbers received a representative exposure to plant conditions prior to the initial inspection, ensuring an adequate basis for operability determination. Subsequent inaccessible snubber inspections will be scheduled in accordance with the existing Technical Specification formula for inspection frequency. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The inaccessible snubbers were inspected following a representative exposure period, and deficiencies were corrected as necessary. In accordance with the Technical Specification the next inspection of inaccessible snubbers will be scheduled based upon the results of the initial inspection. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

In the case of the initial inaccessible snubber inservice inspection period, the nearly three-month period from initial power operation for Waterford 3 sufficiently exercised the snubbers and associated systems to provide a representative "shakedown" period. The proposed change allows LP&L to take credit for the three-month inspection conducted during an outage. While the SRP is silent as to the beginning of the initial inspection period, the three-month inspection is clearly within the guidance of ANSI/ASME OM4–1982. Therefore, the proposed change is similar to example (vi).

As this change requested by the licensee's August 1, 1985 submittal fits the examples provided, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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

Attorney for licensee: Mr. Bruce W. Churchill, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M St., N.W., Washington, D.C. 20036.

NRC Branch Chief: George W. Knighton.

Louisiana Power and Light Company, Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of Amendment Request: August 1, 1985.

Description of Amendment Request: The proposed change would revise the Appendix A Technical Specifications by changing Technical Specification 3/4.9.7 "Crane Travel-Fuel Handling Building" so that use of the spent fuel handling machine is not required for movement of new fuel outside the spent fuel pool.

The purpose of Technical Specification 3/4.9.7 is to restrict movement of loads in excess of the nominal weight of a fuel assembly, control element assembly (CEA), and associated handling tool over other fuel assemblies in the spent fuel pool to ensure that in the event this load is dropped, (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. The original intent of the Specification, as it relates to new fuel, was to require new fuel within the spent fuel pool be handled by the spent fuel handling machine to protect against damage to irradiated fuel.

The proposed change to Technical Specification 3.9.7 will clarify that the use of the spent fuel handling machine is not required for movement of new fuel assemblies outside the spent fuel pool and will also allow for movement of new fuel assemblies in areas other than the spent fuel pool if the spent fuel handling machine is inoperable.

Along this line, the proposed change will bring Technical Specification 3/4.9.7 into conformance with FSAR 9.1.4 which specifies the use of other fuel handling equipment (cask crane, new fuel elevator, etc.) for the movement of new fuel outside the spent fuel pool.

The proposed change consists of the following two parts:

(a) Technical Specification 3.9.7 currently states in part:

Cranes in the fuel handling building shall be restricted as follows: a. The spent fuel handling machine shall be used for the movement of fuel assemblies (with or without CEAs) and shall be OPERABLE with. . . .

The proposed change will add the following note of clarification: Not required for movement of new fuel assemblies outside the spent fuel pool.

(b) The proposed change will add the following Action Statement to Technical Specification 3.9.7:

The provisions of Specification 3.0.4 are not applicable. Specification 3.0.4 normally prevents entry into the applicable mode or condition (movement of fuel assemblies in this case) unless the conditions of the Limiting Condition for Operation are met. This added Action statement will allow for the start of new fuel movement in areas other than the spent fuel pool while Action statement a. is in effect.

Basis for Proposed No Significant Hazards Considerations Determination: The Commission has provided guidance concerning the application of 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 changes are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

The Fuel Handling Accident Analysis in FSAR Chapter 15 is based on the Fuel Handling System described in FSAR Subsection 9.1.4. The proposed change only allows for the use of fuel handling equipment as described by FSAR Subsection 9.1.4 and continues to restrict the movement of heavy loads over fuel assemblies in the spent fuel pool. Therefore, the proposed change will not involve any increase in the probability or consequence of any accident previously evaluated.

Operation of the facility will be in accordance with the assumptions made in the FSAR and the Technical Specification that fuel will be handled in accordance with the designed fuel handling system and movement of heavy loads in the spent fuel pool will be restricted. Therefore, the proposed change will not involve any reduction in the margin of safety.

Operation of the facility will be in accordance with the assumptions made in the FSAR and the Technical Specification that fuel will be handled in accordance with the designed Fuel Handling System and movement of heavy loads in the spent fuel pool will be restricted. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Technical Specification 3/4.9.7 as described in parts a and b above, will allow for the use of fuel handling equipment designed and intended for the movement of new fuel outside the spent fuel pool and bring the Technical Specification into conformance with the FSAR. Therefore, the proposed change is similar to example (vi).

As the change requested by the licensee's August 1, 1985 submittal fits the example provided, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.91; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change, and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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

Attorney for licensee: Mr. Bruce W. Churchill, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M Street, N.W., Washington, D.C. 20036.

NRC Branch Chief: George W. Knighton.

Louisiana Power Light Company, Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of Amendment Request: August 1, 1985.

Description of Amendment Request: The proposed change would revise the Appendix A Technical Specifications by changing Technical Specification 3/4.7.2 "Steam Generator Pressure/ Temperature Limits".

The purpose of Technical Specification 3/4.7.2 is to ensure that steam generator secondary pressure and temperature is limited so that pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The purpose of Specification 3.7.2(b) is to ensure, in the event of a low temperature overpressurization of the steam generator secondary, that an engineering evaluation is completed and it is determined that the steam generator remains acceptable for continued operation prior to increasing its temperature above 115 °F.

The proposed change will allow for steam generator temperatures up to 200 °F prior to completion of the engineering evaluation, consistent with the Revision 3 of the CE Standard Technical Specifications. The present temperature value of 115 °F, with respect to performing an engineering evaluation, is incorrect.

The LIMITING CONDITION FOR OPERATION (LCO) 3.7.2 properly requires that secondary side steam generator temperature be greater than 115 °F when secondary side pressure is above 210 psig. The limitation to 115 °F and 210 psig is based on a steam generator RT_{NDT} of 40 °F, which is sufficient to prevent brittle fracture.

However, in developing the Waterford 3 Technical Specifications, the LCO temperature of 115 °F was inadvertently substituted into ACTION statement 3.7.2.b. As noted above, the CE Standard Technical Specification temperature limitation of 200 °F prior to completion of the engineering evaluation (the ACTION statement temperature) should not have been stated as 115 °F. Raising the Action statement temperature limitation to 200 °F corrects this error and is more conservative in the event of an overpressure condition.

Basis for Proposed No Significant Hazards Considerations Determination: The Commission has provided guidance concerning the application of 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 (i) relates to a purely administrative change to technical specifications, correction of an error, or change in nomenclature.

The lowest service temperature for the secondary side of the steam generators is 115 °F when pressure is 210 psig or greater. Assuming steam generator temperature drops below 115 °F, the Technical Specification as currently written limits temperature to 115 °F or below while an engineering evaluation is performed. In so doing, the Technical Specification unnecessarily exposes the steam generators to the potential for brittle fracture in the event of an overpressure condition. The proposed change would allow an increase in steam generator temperature by to 200 °F while performing the engineering evaluation, thus providing a more conservative condition with respect to brittle fracture should an overpressure condition occur. Therefore, the proposed change will not involve any increase in the probability or consequences of any accident previously evaluated. In fact, the probability of brittle fracture will decrease.

Temperatures less than 200 °F do not impact LOCA or MSLB considerations. The proposed change requires temperatures be maintained to 200 °F or less until it is determined that the steam generator remains acceptable for continued operation. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The lowest service temperature for the secondary side of the steam generators is 115 °F. The Technical Specification, as currently written, limits the temperature to 115 °F or below and is nonconservative because it unnecessarily exposes the steam generator to brittle fracture in the event of an overpressure condition. The proposed change allows for temperatures up to 200 °F, providing for a more conservative condition by allowing temperatures that will place the steam generator material in the ductile range and making them less susceptible to brittle fracture. Therefore, the proposed change will not involve any reduction, but will increase the margin of safety.

The proposed change will change the temperature value of 115 °F by revising Technical Specification 3.7.2(b) to reflect the 200 °F temperature value shown in the CE Standard Technical Specifications, which is the temperature value originally intended for this ACTION. Because the proposed change will correct an error that occurred during development of the Technical Specifications, the proposed change is similar to example (i).

As the change requested by the licensee's August 1, 1985 submittal fits the examples provided, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station or the environment as described in the NRC Final Environmental Statement.

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

Attorney for licensee: Mr. Bruce W. Churchill, Esq., Shaw, Pittman, Potts and Trowbridge, 1800 M St., N.W., Washington, D.C. 20036.

NRC Branch Chief: George W. Knighton.

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

Date of application for amendment: September 22, 1982, as revised June 25, 1984 and May 1, 1985.

Description of amendment request: The proposed amendment will change the Technical Specifications (TS) in the areas of the containment atmosphere control and station battery system. The changes are as follows:

1. Title of TS 3.7/4.7.A.5. is changed from "Oxygen Concentration" to "Containment Atmosphere Control." Technical Specifications and surveillance requirements for the operability of purge and vent valves are added.

2. Appropriate limiting conditions for operation (LCO) and surveillance requirements are added for the new 250 VDC battery installed to supply auxiliary power for the high pressure core injection (HPCI) system.

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 examples (48 FR 14870, April 6, 1983) of actions likely to involve no significant hazards considerations. Example (ii) states "A change that constitutes an additional limitation, restriction or control not presently included in the technical specifications: for example, a more stringent surveillance requirement." The proposed changes fall in this category. Item No. 1 provides additional assurance that the containment purge and vent valves will close as required after an accident and Item No. 2 improves the ability of the plant to cope with severe fires by providing separate 250 VDC power to HPCI and reactor core isolation cooling (RCIC) systems. Therefore, the staff proposes to characterize these as involving no significant hazards considerations.

Local Public Document Room Location: Environmental 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 Chief: Domenic B. Vassallo.

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

Date of application for amendment: August 17, 1984.

Description of amendment request: Item (2) of the proposed amendment request would modify Technical Specification (TS) 5.1.A to more accurately define the property line at the site boundary. Item (1) of the request has already been addressed in Amendment No. 28, dated November 2, 1984.

Basis for proposed no significant hazards consideration determination: The proposed change defines a more up-to-date property line as a result of acquisition of small portion of land at the site boundary. This change does not involve any change in the site boundary. The proposed change is administrative in nature and does not affect the operation of the plant or the safety of the public. For these reasons, the staff concludes that the proposed change would not: (1) Involve a significant increase in the possibility 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 the margin of safety. Therefore, the staff proposes to characterize this as involving no significant hazards consideration.

Local Public Document Room Location: Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota.

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

NRC Branch Chief: Domenic B. Vassallo

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

Date of amendment request: August 27, 1985 (Reference LAR 85-07, Rev. 1).

Description of amendment request: The proposed amendment would revise the Diablo Canyon Units 1 and 2 combined Technical Specifications 3.8.2.1 and 3.8.2.2 and related Bases regarding electrical power systems (battery sets and associated chargers) as follows:

In Specification 3.8.2.1, (a) the Limiting Condition for Operation would be revised to indicate a battery bank is energized from its associated full-capacity charger, and (b) an Action Statement would be added to indicate that with more than one full-capacity charger receiving power simultaneously from a single 480 volt vital bus or any D.C. bus not receiving power from its associated A.C. division, the system is restored to a configuration wherein each charger is powered from its associated 480 volt vital bus within 14 days or the unit is to be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. In Specification 3.8.2.2, Item c, the wording would be revised to clearly indicate that a 125 volt D.C. bus is energized from its associated battery bank, and a full-capacity charger supplied from its associated Operable A.C. vital bus. A statement would be added to the Bases for the Electric Power System to indicate Technical Specification 3.8.2.1, Action c, limits operation to 14 days with an alternate full-capacity charger powered from another 480 volt vital bus.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided guidance concerning the application of standards for determining whether license amendments involve significant hazards considerations by providing certain examples (48 FR 14870). Example (ii) involves a change that constitutes an additional limitation, restriction, or control not presently included in the technical specfications: for example, a more stringent surveillance requirement. The proposed changes fit this example in that (a) the Limiting Condition for Operation would be more restrictive in that it would require that each 125 volt D.C. bus is energized from "its" associated full capacity charger, supplied from "its" associated 480 volt A.C. vital bus, rather than from "an" alternate charger supplied from another vital bus, (b) an additional, restrictive Action Statement would be added to Specification 3.8.2.1 requiring that the battery/charger system be in a configuration wherein each charger is powered from its associated 480 volt bus within 14 days, if the condition of more than one charger receiving power simultaneously from a single vital bus is not rectified, and (c) the supply source (A.C. vital bus) would be added to Specification 3.8.2.2 in accordance with the restrictive changes in (a) and (b) above. Also, clarification would be added to the Bases for the Electric Power System.

The proposed changes are consistent with the NRC Staff position as described in the May 15, 1985, letter from Hugh L. Thompson of the NRC to PG&E regarding the Diablo Canyon Technical Specifications, and subsequent discussions with PG&E. Further justification for the acceptability of operation in the alternate charger alignment for a period of 14 days was provided in PG&E letter DCL-84-214, dated June 14, 1985.

The proposed changes are similar to example (ii) of 48 FR 14870 in that the proposed changes constitute an additional limitation, restriction, or control not presently included in the technical specifications. By revising the Limiting Condition for Operation and adding an Action Statement to the technical specifications, the proposed changes make the technical specifications more restrictive and, therefore, are similar to example (ii) of 48 FR 14870.

On this basis, the NRC proposes to determine that these changes do not involve significant hazards considerations.

Local Public Document Room Location: California Polytechnical State University, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for Licensee: Philip A. Crane, Esq., Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and to Bruce Norton, Esq., Norton, Burke, Berry and French, P.O. Box 10569, Phoenix, Arizona 85064.

NRC Branch Chief: George W. Knighton.

Pacific Gas and Electric Company, Docket No. 50-275 and 50-323, Diablo Canyon Nuclear power Plant, Units 1 and 2, San Luis Obispo County, California

Date of Amendment Request: September 6, 1985 (Reference LAR 85-09).

Description of Amendment Request: The proposed amendment would revise the Diablo Canyon Units 1 and 2 combined Technical Specifications to implement relaxed axial offset control (RAOC) for Unit 1 after 8000 MWD/MTU burnup in Cycle 1. The revision would add Technical Specification 3/4.2.1, "Axial Flux Difference," to include RAOC for unit 1 and would modify the existing Technical Specification 3/4.2.1, "Axial Flux Difference," to be applicable to Unit 2 only. Related Bases information would be added or revised, as appropriate, and administrative changes would be made to make each specification applicable to the appropriate unit.

These changes to implement RAOC would commence at 8000 MWD/MTU for Unit 1 and continue to the end of Cycle 1 for Unit 1 based upon the Westinghouse-performed analysis for Cycle 1. The NRC approved procedure outlined in the Westinghouse report WCAP-10216-PA was used for the analysis, which confirmed that the full range of normal and accident conditions possible with RAOC meets the assumptions of the related safety analysis in the Diablo Canyon FSAR Update.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided guidance concerning the application of standards for determining whether license amendments involve significant hazards considerations by providing certain examples (48 FR 14870). One of the examples of actions involving no significant hazards considerations is example (iv) which involves 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 satifactory way that the criteria have been met. The proposed change fits this example in that it reflects a relaxation in the axial flux difference specification that has been analyzed and found to meet the assumptions of the related safety analysis in the Diablo Canyon FSAR Update. The requested relief is based upon meeting the requirements of WCAP-10216-PA, previously reviewed and approved by the NRC. Thus, the proposed change is similar to example (iv) of 48 FR 14870 of actions not likely to involve a significant hazards consideration.

Local Public Document Room Location: California Polytechnical State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for Licensee: Philip A. Crane, Esq., Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and to Bruce Norton, Esq., Norton, Burke, Berry and French, P.O. Box 10569, Phoenix, Arizona 95064.

NRC Branch Chief: George W. Knighton.

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

Date of Amendment Request: September 20, 1985 (Reference LAR 85-10).

Description of Amendment Request: The proposed amendment would revise the Diablo Canyon combined Technical Specifications for Units 1 and 2 to allow performance of the first inservice snubber visual inspection for Unit 2 following completion of the power ascension program. Technical Specification 4.7.7.1b presently requires the inspection to be performed after 4 months but within 10 months of commencing Power Operation. The change requested would revise Technical Specification 4.7.7.1b to allow performance of the first inspection after completion of the power ascension test program or after four months, but within ten months, of commencing Power Operation. As defined in the Diablo Canyon Technical Specifications, "Power Operation" is operation at a power level greater than five percent of rated thermal power.

Power Operation of Unit 2 is presently targeted for early October 1985. The power ascension test program is scheduled for approximately 12 weeks to be followed by the strainer removal outage. PG&E desires to perform the snubber visual inspection of Technical Specification 4.7.7.1b during the outage following the power ascension test program.

Although this change would revise the Units 1 and 2 combined Technical Specifications, it only affects Unit 2 since the first inservice visual inspection for Unit 1 has been completed.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided guidance concerning the application of 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 changes are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

PG&E has already performed a Unit 2 snubber walkdown and visual inspection for all mechanical snubbers following plant heatup in accordance with IE Bulletin 81-01. Minor problems found during the inspection were corrected and the snubbers were reinspected. The Unit 2 will experience load swings, trips, and other transients during the approximately three-month long power ascension test program that will cause movement of snubbers typical of that expected throughout the life of the plant. Therefore, a snubber inspection at the conclusion of the power ascension testing and trip from 100% power is appropriate. No additional information on snubber performance would be gained by delaying the visual inspection of snubbers for one additional month while the unit is in steady state commercial operation (as would be required under the current Technical Specification).

The proposed amendment is designed to allow performance of the first inservice snubber visual inspection for Unit 2 during the outage following the power ascension test program. This change would not necessitate physical alteration of the plant or changes in parameters governing normal plant operation and would provide adequate information on snubber operability. The proposed change is also in conformance with ANSI/ASME Standard OM4-1982, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)". Section 3.2.3 of the standard, Inservice Examination Frequency, states: "The initial inservice examination of all snubbers shall be initiated after at least 2 months of power operation and shall be completed prior to 12 calendar months after initial criticality."

The inspection will be performed after Unit 2 has been subjected to an acceptable number of plant transients. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated and will not involve a significant reduction in a margin of safety. Accordingly, the proposed change is similar example (vi).

Therefore, the staff proposes to determine that performance of snubber surveillance in accordance with the proposed revision does 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. Accordingly, the Commission proposes to determine that this change involves no significant hazards considerations as defined by 10 CFR 50.92.

Local Public Document Room Location: California Polytechnical State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorneys for Licensee: Philip A. Crane, Esq., Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120 and to Bruce Norton, Esq., Norton, Burke, Berry and French, P.O. Box 10569, Phoenix, Arizona 95064.

NRC Branch Chief: George W. Knighton.

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

Date of amendment request: July 31, 1985 as supplemented on September 13, 1985.

Description of amendment request: In January 1984 the licensee experienced ice formation on the spray nozzles of the spray pond. On August 31, 1984 the licensee provided to the staff a long-term solution to preclude freezing problems in the spray pond. The licensee's proposed solution would add an automatic start capability to the recently installed self-priming pumping system. This modification will allow draindown of the spray arrays without operator action. A new motor operated valve will be installed in each spray array drain line to isolate the spray arrays from the drain pumps. These new drain valves will be interlocked with the drain pumps and riser level monitoring instrumentation to allow automatic pumpdown of the spray risers.

This plant modification is reflected in a proposed change to Table 3.8.4.2-1 of the Technical Specifications for both Units 1 and 2. The licensee has proposed to add these valves to Table 3.8.4.2-1 (MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION). These valves are safety related and the valves have thermal overload protection devices; however, this protection is continuously bypassed except during testing. By design all safety related valves have their thermal overload protection devices continuously bypassed except during testing so that the valves can perform their safety related function beyond that which the thermal overload protection would limit

Basis for Proposed No Significant Hazards Consideration Determination: The licensee in his letter dated July 31, 1985, as supplemented on September 13, 1985 stated that:

(1) The proposed change does not involve a significant increase in the

probability or consequences of an accident previously evaluated. Neither the drain pumps nor the level detection system are safety related since these systems are used only to maintain the spray arrays in an operable condition. The drain valves provide a boundary between the ASME Section III Residual Heat Removal Service Water (RHRSW) Emergency Service Water (ESW) piping and the non-quality drain pumps and are safety related. The safety function of the drain valves is to close when the spray array isolation valves open and an interlock is provided that prevents the drain valves from opening unless the spray array isolation valves are 100% closed.

The drain valves are designed to ASME Section III Class 2 and are Seismic Category I. The motor operators are Class 1E and are powered from existing Class 1E motor control centers. Since the level instrumentation system is non-Class 1E, proper separation between Class 1E and non-Class 1E circuits is provided.

A fire will not jeopardize the safe shutdown of the plant due to the installation of the automatic drain system. This modification was analyzed with respect to fire protection and was found to be consistent with the Fire Hazards Analysis for the plant.

The proposed modification will allow automatic pumpdown of the spray arrays, thereby providing protection against freezing. This decreases the dependency on operators and thus contributes to safety. This modification does not jeopardize the capability of the spray arrays, ESW or RHRSW of performing its intended safety functions. Therefore, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment related to safety as previously evaluated.

(2) The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed action does not alter the function or operation of any safety related systems. This change does not compromise separation criteria nor does it allow a single failure to prevent any safety related systems from performing their intended safety functions. This design is consistent with the design philosophy as described in the FSAR and does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR.

(3) The proposed change does not involve a significant reduction in a margin of safety, since this modification does not affect the ability of the Spray Pond, ESW or RHRSW to provide sufficient cooling nor does it affect the redundancy of these systems.

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. In addition, the Commission has provided guidance concerning the application of the no significant hazards consideration standards by providing certain examples (48 FR 14870). One of the examples of actions not likely to involve a significant hazards consideration, example (ii), is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications: for example, a more stringent surveillance requirement. Since the licensee has proposed to add valves subject to controls and requirements to the Technical Specifications, this change is encompassed by the example (ii). Based on the above, the staff proposes to find that this change does not involve a significant hazards consideration.

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, N.W., Washington, D.C. 20036.

NRC Branch Chief: W. Butler.

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

Date of amendment request: August 1, 1983 as revised October 26, 1983.

Description of amendment request: Amendment 11 to Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant was issued July 30, 1985, and addressed a majority of the proposed Technical Specifications (TS) changes requested in the August 1, 1983 submittal. A portion of the proposed changes was not covered by the initial notice published in the **Federal Register** on November 22, 1983 (50 FR 52824). In the letter dated August 1, 1983, Rochester Gas and Electric (RG&E) proposed that the Ginna TS 4.6.2.e be added, requiring the performance of a battery discharge test at least once each 60 months. In a second letter dated October 26, 1983, RG&E proposed that the Ginna TS 4.6.2.f be added, requiring the battery discharge test to be performed annually for any battery that shows degradation. Degradation is indicated when the battery capacity drops more than 10% of rated capacity for its average on previous discharge tests, or is below 90% of the manufacturer's rating.

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). Example (ii) of actions not likely to involve a significant hazards consideration is a change that constitutes an additional restriction or control not presently included in the TS. Both of the proposed changes are a result of the Systematic Evaluation Program (SEP) for the R.E. Ginna Nuclear Power Plant. Each of the changes introduces an additional restriction or control which does not currently exist. Because the proposed addition of TS 4.6.2.e and 4.6.2.f is encompassed by example (ii), the staff proposes to determine that the requested action does not involve a significant hazards consideration.

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

Attorney for licensee: Harry H. Voigt, Esquire, LaBoeuf, Lamb, Leiby and MacRae, 1333 New Hampshire Avenue, N.W., Suite 1100, Washington, D.C. 20036.

NRC Branch Chief: John A. Zwolinski.

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 amendment request: October 8, 1982; April 26, 1984; August 2, 1985; September 25, 1985.

Description of amendment request: The amendment would revise Technical Specification (T.S.) Table 3.6-1, "Containment Isolation Valves," and bases section 3/4.6.1.2, "Containment Leakage." Two valves are being deleted from T.S. Table 3.6-1 because they are going to be removed from the plant and their lines capped. Eight valves listed in T.S. Table 3.6-1 will be footnoted to indicate that they are not subject to Type C leak tests. Also, the bases section is being changed to clarify that conservatism exists in the methods to demonstrate a water seal.

Basis for proposed no significant hazards consideration determination: The two valves being removed will have their lines capped. Those caps will ensure containment isolation better than the two valves provided. The eight valves being footnoted to indicate that they do not require Type C leak tests will remain sealed with water during a loss of coolant accident and do not constitute potential containment atmosphere leak paths. This is consistent with 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing For Water-Cooled Power Reactors," which does not require Type C, leak tests for valves that will remain sealed with water during a loss of coolant accident. Finally, the bases is being changed to clarify that methods used to demonstrate water seals are conservative. The Commission has provided certain examples (48 FR 14870) of actions likely to involve no significant hazards considerations. The request involved in this case does not match any of those examples. However, the staff has reviewed the licensee's request for the above amendment and determined that should this request be implemented, it will not (1) involve a significant increase in the probability or consequences of an accident previously evaluated because a loss-of-coolant accident is not made more probable, the caps will be better containment isolation than the two valves, and the eight valves will have water seals that they do not consitute potential containment atmosphere leak paths.

Also, it will not (2) create the possibility of a new or different kind of accident from any accident previously evaluated because the closed valves that are being changed to pipe caps never have to be open during plant operation.

Finally, it will not (3) involve a significant reduction in a margin of safety because the pipe caps and water seals will maintain effective containment isolation in case of the design basis loss-of-coolant accident. Accordingly, the Commission proposes to determine that this change does not involve significant hazards considerations.

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

Attorney for licensee: Randolph L. Mahan, South Carolina Electric and Gas Company, P.O. Box 764, Columbia, South Carolina 28218.

NRC Branch Chief: Elinor G. Adensam.

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 requests: January 25, 1984 and August 20, 1985 (Reference PCN-91); May 23, 1984, August 7, 1984 and August 20, 1985 (Reference PCN-137).

Description of amendment request: The proposed changes would revise Technical Specifications 3/4.8.1.1, "Electrical Power Systems—A.C. Sources—Operating," and 3/4.8.1.2, "Electrical Power Systems—A.C. Sources—Shutdown," as follows: 1) PCN-91 would delete Technical Specification 4.8.1.1.1.d.6, a diesel generator surveillance requirement, to test reloading of a diesel generator following its failure with offsite power not available, consistent with the recommendation of Generic Letter 83-30; 2) PCN-137 would revise Technical Specification 3/4.8.1.2 to include only those limiting conditions for operation (LCO's) and surveillance requirements which directly relate to the operability of the A.C. Power sources required under shutdown and refueling conditions.

Basis for Proposed No Significant Hazards Determination: The Commission has provided guidance concerning the application of 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 in the probability or consequences of a previously-analyzed accident or may reduce in some way a safey margin, but where the results of the change are clearly within all acceptance 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. Each proposed change discussed below is similar to Example (vi) of 48 FR 14870. Therefore, it is proposed that these changes do not involve significant hazards considerations. The following is a description of each proposed change to the technical specifications and a discussion of how each change is similar to Example (vi) of 48 FR 14870.

Specific Changes Requested and Basis for Proposed No Significant Hazards Determination: 1. Proposed Change PCN-91.

The proposed change would delete Surveillance Requirement 4.8.1.1.2.d.6 of Technical Specification 3/4.8.1, "A.C. Sources," which defines the operability requirements for A.C. electrical power sources. T.S. 4.8.1.1.2 states the requirements for demonstrating diesel generator operability. Surveillance Requirement 4.8.1.1.2.d.6 states that once every eighteen months, during shutdown, loss of both offsite and diesel generator power must be simulated in order to verify that in this situation all loads depending on the diesel generators will be shed and the diesels will be reloaded in accordance with design requirements. The proposed change would delete this surveillance requirement.

The proposed change is similar to Example (vi) of 48 FR 14870 in that it relates to a change that 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. Generic Letter No. 83-30, "Deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 is based on its inconsistency with 10 CFR 50, Appendix A, General Design Criterion 17, "Electrical Power Systems," Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," and the Standard Review Plan Sections 8.2, "Offsite Power System," and 8.3.1, "A.C. Power Systems (Onsite)." These references, which delineate the requirements for diesel generators, do not require diesel generator operability tests such as that currently specified by T.S. 4.8.1.1.2.d.6. Because the result of this change would make the technical specifications conform with all acceptance criteria, it is similar to Example (vi) of 48 FR 14870.

2. Proposed Change PCN-137.

The proposed change would revise Technical Specification 3/4.8.1.2, "Electrical Power Systems—A.C. Sources—Shutdown," which defines the requirements for A.C. electrical power source operability during operating Modes 5 and 6. The surveillance requirements governing emergency diesel generator (EDG) operability in Modes 5 and 6 currently prescribe all those surveillances required in Modes 1 through 4 with one exception. The proposed change would revise T.S. 3/4.8.1.2 and T.S. Bases 3/4.8 to include only those limiting conditions for operation and surveillance requirements which verify operability of the A.C. sources required under shutdown and refueling conditions (Modes 5 and 6, respectively). The following functions are not required to be performed by the EDG during Modes 5 and 6 and, on that basis the surveillance requirements relating to these functions would be deleted by the proposed change. The items to be deleted are: 1) automatic start of the EDG on an emergency safety features (ESF) signal, on loss of offsite power in conjunction with an ESF signal, or from a test mode; and 2) automatic load sequencing on a ESF signal. Also proposed to be deleted is the surveillance requirement specifying the maximum auto-connected loads applicable in Modes 1, through 4, since in Modes 5 and 6 no loads except the permanently connected shutdown loads are automatically connected to the EDG. In addition, it is proposed that the specification stating the minimum volume of diesel generator fuel to be stored be revised to require a minimum of 37,600 gallons of fuel rather than 47,000 gallons of fuel.

The proposed change is similar to Example (vi) of 48 FR 14870 in that it may result in some increase in 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 acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

SRP Section 8.3.1, "A-C Power Systems (Onsite)," delineates the acceptance criteria regarding A.C. electrical power sources. For specific guidelines it references Regulatory Guides 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electrical Power Systems at Nuclear Power Plants," and Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators." Regulatory Guide 1.108 states that diesel generator design should include provisions so that the testing of the units will simulate the parameters of operations that would be expected if actual demand were to be placed on the system. The first part of the proposed change revises the T.S. 3/4.8.1.2 surveillance requirements to more accurately reflect the parameters of operation that would be expected if an actual demand were to be placed on the diesel generator with the plant in cold shutdown or refueling modes. Regulatory Guide 1.137 states that the calculation of fuel-oil storage requirements may be based on the time-dependent loads of the diesel generator. For this calculational method, the minimum required capacity should include the capacity to power the engineered safety features. The second part of the proposed change reduces the minimum required volume of fuel storage system fuel for operation in Modes 5 and 6. The largest anticipated load in Mode 5 and 6 (considering all loads required to mitigate the consequences of the range of postulated accidents and all loads which facilitate plant operation maintenance) has been calculated to be less than 80% of the EDG full rated capacity. Therefore, in accordance with Regulatory Guide 1.137, less fuel is required to be stored during Modes 5 and 6 operation since the maximum diesel generator load during these modes is only 80% of full rated capacity. This part of the proposed change is also consistent with Regulatory Guide 1.108 since it more accurately reflects the parameters of operation (i.e., operation in Modes 5 and 6 only) specified for this technical specification.

Based on the above discussion, the NRC staff proposed to determine that these changes meet the SRP acceptance criteria and are similar to Example (vi) of 48 FR 14870.

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

Attorney for licensees: 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.

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 amendment requests: February 14, 1979, as supplemented September 21, 1982 and August 30, 1985.

Description of amendment requests: The amendment revises Technical Specifications (TS) 4.0, 4.1, 4.2, 4.3, 4.5, 4.7, 4.8 and 4.11 to add Surveillance Requirements to ensure that inservice testing of ASME Code Class 1, 2 and 3 pumps and valves and inservice inspection of ASME Code Class 1, 2 and 3 components will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda. The amendment request was initially noticed on August 23, 1983 (48 FR 38428). By letters dated February 14, 1979 and September 21, 1982, Virginia Electric and Power Company submitted proposed license amendments for NRC review and approval which reflected changes to the surveillance requirements.

This notice includes changes requested in a subsequent submittal dated August 30, 1985. This submittal updates the previous submittals and provides supplemental information and clarification as requested by the staff's May 28, 1985 request for additional information.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of these standards by providing examples (48 FR 14870). One of these examples relates to changes which constitute an additional limitation, restriction, or control. The licensee has submitted an updated version of the Inservice Inspection and Testing Program for Units 1 and 2. The Technical Specification changes are requested to ensure that the revised program is in accordance with the applicable ASME Code and Addenda as required by 10 CFR 50.55, "Codes and Standards." Since the proposed changes add requirements to ensure compliance with the regulations, these changes fall within example (ii) of actions not likely to involve significant hazards considerations. On this basis, the staff proposes to determine that the application does not involve a significant hazards consideration.

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

Attorney for licensee: Mr. Michael W. Maupin, Hunton and Williams, Post Office Box 1535, Richmond, Virginia 23213.

NRC Branch Chief: Steven A. Varga.

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

Date of amendment request: August 30, 1985.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to: (1) Correct typographical errors and make clarifications; (2) remove reference to three loop operation; (3) revise the maximum allowable core inlet temperature; (4) revise the Linear Heat Generation Rate (LHGR) limit; (5) revise the control-rod-motion-related peaking multipliers that are applied to measured LHGR for comparison to the LOCA limit; (6) modify the method for combining the independent uncertainty parameters applied to the measured LHGR, and (7) modify the Safety Injection Acutation Signal (SIAS) setpoint.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a no significant hazards consideration exists by providing certain examples (48 FR 14870). The examples include: (i) A purely administrative change to the TS to achieve consistency throughout the TS, correct errors or to change nomenclature; and (iii) 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, and assuming 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.

Item (1) is encompassed by example (i) of actions not likely to involve a significant hazards consideration.

Minor changes to the code and code assumptions for performing LOCA analyses have resulted in proposed changes to the TS. In addition, the results of the core reload analyses have resulted in additional proposed changes to the TS. Items (4), (5) and (6) above are encompassed by example (iii) of the Commission's examples of amendments not likely to involve a significant hazards consideration. Item (4) proposes a revised LHGR limit based on the core reload analysis that takes into account worst-case axial power shapes to demonstrate compliance with Appendix K to 10 CFR Part 50. Item (5) proposes to modify peaking multipliers related to control rod motion, based on revised analyses for LHGR. Item (6) proposes to modify the method identified in the TS for combining the uncertainty parameters associated with determining LHGR. The method is being changed from a multiplicative to a statistical combination of the uncertainty parameters.

The staff has reviewed Items (2), (3) and (7) 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 identified, 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 to safety. The basis for this determination follows:

Item (3) proposes to increase the TS maximum allowable core inlet temperature by 5 °F, from 515 °F to 520 °Φ, to allow for increased flexibility in future plant operations. In addition, the licensee proposes in item (7) to reduce the SIAS setpoint from 1700 psig to 1650 psig to reduce the likelihood of inadvertent safety injection actuations following a reactor trip (i.e., an unnecessary challenge to safety equipment). The core reload analysis shows that modifying these two TS parameters has a minimal effect on the consequences of accidents previously evaluated, and the margin to thermal limits remain well within applicable acceptance criteria.

Item (2) of the licensee's submittal proposes to remove a reference to three-loop operating parameters from one of the TS tables. Operation of Yankee with three loops is not allowed, and removal of references to three loop operations will make the TS consistent with allowed operating conditions.

Based on the above discussions, the staff proposes to determine 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.

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.

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

Date of amendment request: August 16, 1985 as revised on September 24, 1985.

Description of amendment request: The proposed amendment accommodates the reactor core reload I1 fuel design.

Date of publication of individual notice in **Federal Register**: October 1, 1985 (50 FR 40076).

Expiration date of individual notice: October 31, 1985.

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

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, N.W., 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.

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

Date of application for amendment: June 18, 1985.

Brief description of amendment: The amendment changes the Technical

Specifications by changing the Reactor Low Water Level (inside shroud) trip requirement to "greater than or equal to 307 inches above vessel zero (approximately ⅔ core height)."

Date of issuance: October 9, 1985.

Effective date: 30 days after issuance.

Amendment No.: 90.

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

Date of initial notice in **Federal Register**: August 14, 1985 (50 FR 32788)

The Commission's related evaluation of the amendment is contained in Safety Evaluation dated October 9, 1985.

No significant hazards consideration comments received: No.

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

Commonwealth Edison Company, Docket No. STN 50-454, Byron Station, Unit 1, Ogle County, Illinois

Date of amendment request: June 28, 1985.

Brief description of amendment: The amendment approves Technical Specification changes relating to administrative controls for access to high radiation areas during certain emergency situations and replaces a page inadvertently omitted in the printing of the Technical Specifications.

Date of issuance: October 1, 1985.

Effective date: October 1, 1985.

Amendment No.: 1.

Facility Operating License No. NPT-37. Amendment revised the Technical Specification.

Date of initial notice in **Federal Register**: July 31, 1985 (50 FR 31067).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation, dated October 1, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Rockford Public Library, 215 N. Wyman Street, Rockford, Illinois 61103.

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

Date of amendments request: April 17, 1985.

Brief Description of amendments: The proposed amendments to Operating License NPF-11 and Operating License NPF-18 would the La Salle Units 1 and 2 Technical Specifications to incorporate the following: (1) Correction of typographical and administrative errors and inclusion of a limit curve when the end-of-cycle reactor pump trip is inoperable; (2) a statement that Specification 3.0.4 does not apply in Specification 3.6.3 by permitting reactor startup as long as assurance is provided that a system inoperable would not affect plant safety; (3) clarification to indicate required action on failure of either "Full In" or "Full Out" reactivity position and specifying system surveillance of "Full In" indication; (4) correction allowed for time decay of liquid effluent batch releases for lower limit of detection; (5) new method of calculating the kilowatt capacity for electric heaters in the control room emergency air make-up train; and (6) the reactor core isolation cooling differential temperature instrumentation with respect to set points surveillance requirements and required remedial actions.

Date of issuance: October 2, 1985.

Effective date: October 2, 1985.

Amendment Nos.: 28 and 14.

Facility Operating Licenses No. NPF-11 and NPF-18 Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 4, 1985 (50 FR 107)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Public Library of Illinois Valley Community College, Rural Route No. 1, Oglesby, Illinois 61348

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

Date of application for amendment: April 10, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to remove the requirement of waiting 400 continuous hours after shutdown before unloading more than one region of fuel assemblies. The amendment permits the discharge of the entire reactor core after a continuous interval of 131 hours following shutdown, the current time constraint for movement of only one region of fuel assemblies.

Date of Issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 98.

Facilities Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: May 21, 1985 (50 FR 20973)

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

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

Date of application for amendment: June 18, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to limit overtime for critical shift job positions, changes the audit frequency of the Emergency Preparedness Program and Safeguards Contingency Plan, and clarifies the Quality Assurance Record retention requirements.

Date of issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 97.

Facilities Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: August 28, 1985 (50 FR 34936)

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

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

Date of application for amendment: July 31, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to permit a one- time extension of the surveillance interval limits for various systems and components. Specifically the Technical Specifications are modified to extend the 3.25 total time interval limit over three consecutive surveillance intervals to allow testing to be performed during the scheduled 1986 refueling/ maintenance outage rather than requiring a special plant shutdown solely to perform these tests

Date of issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 99.

Facilities Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 28, 1985 (50 FR 34937)

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

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

Date of application for amendment: November 14, 1984, as revised on January 17, 1985.

Brief description of amendment: The amendment modifies the Big Rock Point Technical Specifications by implementing a definition of operability and incorporating Limiting Conditions for Operation of redundant safety systems.

Date of issuance: October 2, 1985.

Effective date: October 2, 1985.

Amendment No.: 78.

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

Date of initial notice in Federal Register: May 21, 1985 (50 FR 20974).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1985.

No significant hazards consideration comments received: No.

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

Dairyland Power Cooperative, Docket No. 50–409, La Crosse Boiling Water Reactor, Vernon County, Wisconsin

Date of application for amendment: March 21, 1984.

Brief description of amendment: The amendment adds a new paragraph 4.2.23 to the Technical Specification to require the Demineralized Virgin Water Tank to be operable with a minimum water level of 1 foot. In addition the amendment adds a surveillance requirement to verify the minimum water level in the tank at least once per 7 days, and adds a basis for the above requirements.

Date of Issuance: October 8, 1985.

Effective date: October 8, 1985.

Provisional Operating License No. DPR–45. Amendment revised the Appendix A Technical Specifications.

Date of initial notice in **Federal Register:** May 23, 1984 (49 FR 21829).

The Commission's related evaluation for the license amendment is contained in a Safety Evaluation dated October 8, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601.

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

Date of application for amendments: April 9, 1985.

Brief description of amendments: The amendments change Technical Specification surveillance requirements related to the inservice inspection program for snubbers.

Date of issuance: September 30, 1985.

Effective date; September 30, 1985.

Amendment Nos. 46 and 27.

Facility Operating License Nos. NPF–9 and NPF–17. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 28, 1985 (50 FR 34939).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: May 31, 1985.

Brief description of amendments: These amendments revise the Station's common Technical Specifications (TSs) to support the operation of Oconee Unit 3 at full rated power during the upcoming Cycle 9. The amendments change the following areas: 1) Core Protection Safety Limits (TS 2.1); 2) Protective System Maximum Allowable Setpoints (TS 2.3); 3) Rod Position limits (TS 3.5.2); and 4) Power Imbalance Limits (TS 3.5.2).

Date of issuance: September 19, 1985.

Effective date: September 19, 1985.

Amendments Nos.: 142, 142, 139.

Facility Operating Licenses Nos. DPR–38, DPR–47 and DPR–55. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** July 17, 1985 (50 FR 29009).

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

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: February 10, 1983.

Brief description of amendments: These amendments revise the Station's common Technical Specifications (TSs) to allow the use of the Reactor Coolant System (RCS) inservice leak and hydrostatic test heatup and cooldown limitations during the performance of leak tests of connected systems when the RCS pressure-temperature limits are controlling.

Date of issuance: October 9, 1985.

Effective date: October 9, 1985.

Amendments Nos.: 143, 143 and 140.

Facility Operating Licenses Nos. DPR–38, DPR–47 and DPR–55. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1983 (48 FR 56502).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 9, 1985

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: July 25, 1984.

Brief description of amendment: This amendment deletes Surveillance Requirement 4.8.1.1.1.a.2 which requires that the operability of the sump pumps in the tunnel containing the DC control supply to the 230kv switchgear be verified at least once per seven days.

Date of issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 83.

Facility Operating License No. DPR–72. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 21, 1984 (49 FR 45949).

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

No significant hazards consideration comments received: No.

Local Public Document Room Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida.

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

Date of application for amendment: September 28, 1984.

Brief description of amendment: The amendment corrects errors and inconsistencies and clarifies certain radiological effluent Technical Specifications.

Date of issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 84.

Facility Operating License No. DPR-72. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 21, 1984 (49 FR 45948).

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

No significant hazards consideration comments received: No.

Local Public Document Room Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida.

Iowa Electric Light and Power Company, Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: August 31, 1984, as supplemented September 13, 1985.

Brief description of amendment: This amendment revises the Technical Specifications to incorporate restrictions required by NUREG-0737 Item I.A.1.3.1, regarding overtime for plant operators.

Date of issuance: October 10, 1985.

Effective date: October 10, 1985.

Amendment No.: 126.

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

Date of initial notice in **Federal Register:** October 24, 1984 (48 FR 42827).

Subsequent to the initial notice, the licensee, by a letter dated September 13, 1985, clarified the wording of the Technical Specification change and made it clearly consistent with the description of the requested change as described in the August 31, 1984 application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station Unit No. 2, Town of Waterford, Connecticut

Date of application for amendment: July 15, 1985.

Brief description of amendment: This amendment corrects a typographical error on Figure 3.2-2a of the Technical Specifications.

Date of issuance: October 3, 1985.

Effective date: October 3, 1985.

Amendment No.: 105.

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

Date of initial notice in **Federal Register:** August 14, 1985 (50 FR 32787 at 32798).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1985.

No significant hazards consideration comments received: No.

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

Local Public Document Room location: Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut.

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

Date of application for amendment: May 29, 1985.

Brief description of amendment: This amendment deletes Appendix B in its entirety and provides new Appendix A Technical Specifications sections defining limiting conditions for operation and surveillance of radioactive effluents, concentration and treatment and total dose.

Date of issuance: October 1, 1985.

Effective date: January 1, 1986.

Amendment No. 106.

Provisional Operating License No. DPR-21. This amendment revised the Technical Specifications related to radioactive waste management, i.e., Radiological Environmental Technical Specifications, and the license.

Date of initial notice in **Federal Register:** August 14, 1985 (50 FR 32798).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

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

Date of application for amendment: February 15, 1983.

Brief description of amendment: The amendment revises the Technical Specifications Section 3.13/4-13, "Fire Suppression Water Systems" to change the term "screen wash pump" to "screen wash/fire pump" and reword the bases accordingly.

Date of issuance: October 7, 1985.

Effective date: October 7, 1985.

Amendment No.: 33.

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

Date of initial notice in **Federal Register:** June 20, 1984 (49 FR 25365).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 7, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: April 10 and June 14, 1985.

Brief description of amendment: The amendment revises the Technical Specifications by raising the K-effective limit on the spent fuel storage pool from 0.90 to 0.95 and that the infinite multiplication factor be less than or equal to 1.31 for the new fuel assemblies and 1.33 for the spent fuel assemblies.

Date of issuance: October 8, 1985.

Effective date: October 8, 1985.

Amendment No.: 34.

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

Date of initial notice in **Federal Register:** August 14, 1985 (50 FR 32799).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 8, 1985.

No significant hazards consideration comments received: No.

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

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: April 19, 1984, as supplemented October 2, 1984.

Brief description of amendments: These amendments correct errors and establish consistency in the reactor water level setpoint values, lower the main steam line isolation valve low water isolation setpoint from low-low to low-low-low, and revise the audit frequency of the Facility Emergency Plan and implementing procedures to conform with the Commission's regulations.

Date of issuance: October 2, 1985.

Effective date: October 2, 1985.

Amendments Nos.: 111 and 115.

Facility Operating Licenses Nos. DPR-44 and DPR-56. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 24, 1984 (49 FR 29918).

Since the initial notice, the licensee supplemented the application by letter dated October 2, 1984. This submittal provided additional information concerning this amendment request as a result of certain concerns expressed by the NRC staff to the licensee during its review. This submittal did not affect the requested changes proposed in the original application dated April 19, 1984.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation October 2, 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.

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

Date of application for amendment: January 29, 1985, revised June 14, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to reduce the frequency of diesel generator testing and allow the engine to be warmed up for most tests before increasing speed. The test starts from ambient conditions are to be conducted semi-annually instead of monthly. NRC letter 84-15 identified that cold fast starts of diesel generator sets contribute to premature diesel engine degradation and excessive diesel generator testing contributes to unnecessary wear.

Date of issuance: October 4, 1985.

Effective date: October 4, 1985.

Amendment No.: 107.

Facility Operating License No. NPF-1. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: March 27, 1985 (50 FR 12132 at 12159).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1985.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: March 16, 1979, as supplemented December 12, 1979, February 19, 1985, and April 24, 1985.

Brief description of amendment: The amendment revises the Technical Specifications to provide conformance with the Commission's regulations governing Inservice Inspection as set forth in 10 CFR 50.55a(g). It also revises the Technical Specifications governing inspection of steam generator tubes.

Date of issuance: September 30, 1985.

Effective date: September 30, 1985.

Amendment No.: 76.

Facility Operating License No. DPR-54. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 21, 1983 (48 FR 56509) and May 21, 1985 (50 FR 20988).

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

No significant hazards consideration comments received: No.

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

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

Date of amendment request: July 10, 1985, as supplemented by letter dated August 9, 1985.

Description of amendment request: The amendment extends the initial 18-month surveillance interval for manual initiations of the reactor trip system and engineered safety features actuation system (ESFAS), portions of diesel generator testing, ESFAS actuations on safety injection and loss of offsite power, containment spray actuation testing, Phase A and B containment isolations, and Class 1E battery service tests.

Date of issuance: October 3, 1985.

Effective date: October 3, 1985.

Amendment No.: 8.

Facility Operating License No. NPF-30: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 3, 1985 (50 FR 35628).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1985.

No significant hazards consideration comments received: No.

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.

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

Date of application for amendments: June 4, 1976 as modified January 28, 1980, October 7, 1983, December 20, 1984 and April 12, 1985.

Brief description of amendments: The amendments added limiting conditions for operation and surveillance requirements for monitoring liquid and gaseous radiological effluents. Additional environmental sampling locations have been added and additional managerial review responsibilities and reporting requirements have been added relating to radioactive releases.

Date of issuance: October 3, 1985.

Effective date: 20 days from date of issuance.

Amendment No.: 97 and 101.

Facility Operating License No. DPR-24 and DPR-27: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 23, 1983 (48 FR 38382 at 38430) Renoticed November 22, 1983 (48 FR 52804 at 52840) Renoticed February 27, 1985 (50 FR 7979 at 8011) Renoticed July 31, 1985 (50 FR 31061 at 31076).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 1985.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Joseph P. Mann, Library, 1516 Sixteenth Street, Two Rivers, Wisconsin.

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

Date of amendment request: April 17, 1984, as supplemented August 7, 1984, and revised April 5, 1985.

Description of amendment request: (1) Revise the technical specification (TS) Bases for Pressurizer Code Safety valve capacity (2) administratively removes snubbers no longer required after replacement of pressurizer code safety valves, (3) adds TS for reactor coolant system vents, and (4) adds TS for Degraded Grid voltage system.

Date of issuance: October 1, 1985.

Effective date: October 1, 1985.

Amendment No.: 84.

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

Date of initial notice in **Federal Register:** May 14, 1984 (49 FR 20391).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1985.

No significant hazards consideration comments received: No.

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

Dated at Bethesda, Maryland this 16th day of October 1985.

For the Nuclear Regulatory Commission.

Edward J. Butcher,

Acting Chief Operating Reactors Branch No. 3, Division of Licensing.

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