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DOCKET NO(S). 50-382

Mr. R. S. Leddick
Vice President - Nuclear Operation
Louisiana Power & Light Company
142 Delaronde Street
New Orleans, Louisiana 70174

SUBJECT: LOUISIANA POWER & LIGHT COMPANY -
WATERFORD STEAM ELECTRIC STATION, UNIT 3

The following documents concerning our review of the subject facility are transmitted for your information.

- Notice of Receipt of Application, dated _____.
- Draft/Final Environmental Statment, dated _____.
- Notice of Availability of Draft/Final Environmental Statement, dated _____.
- Safety Evaluation Report, or Supplement No. _____, dated _____.
- Notice of Hearing on Application for Construction Permit, dated _____.
- Notice of Consideration of Issuance of Facility Operating License, dated _____.
- ~~XXXXXX~~ **Bi-Weekly** Notice; Applications and Amendments to Operating Licenses Involving no Significant Hazards Considerations, dated 2/12/86 (See page 5283).
- Application and Safety Analysis Report, Volume _____.
- Amendment No. _____ to Application/SAR dated _____.
- Construction Permit No. CPPR- _____, Amendment No. _____ dated _____.
- Facility Operating License No. _____, Amendment No. _____, dated _____.
- Order Extending Construction Completion Date, dated _____.
- Other (Specify) _____

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

OFFICE	PBD7						
SURNAME	JLee						
DATE	2/18/86						



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

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Vice President - Nuclear Operation
Louisiana Power & Light Company
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Bi-Weekly
- ~~Monthly~~ Notice; Applications and Amendments to Operating Licenses Involving no Significant Hazards Considerations, dated 2/12/86 (See page 5283).
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- Other (*Specify*) _____

Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See next page

Mr. R. S. Leddick
Louisiana Power & Light Company

Waterford 3

cc:

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Louisiana Public Service Commission
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Baton Rouge, Louisiana 70804

Mr. William H. Spell, Administrator
Nuclear Energy Division
Office of Environmental Affairs
P. O. Box 14690
Baton Route, Louisiana 70898

President, Police Jury
St. Charles Parrish
Hahnville, Louisiana 70057

files, 1950-65, and correspondence, memoranda, outlines, and reports relating to research activities, 1942-52.

4. Department of the Air Force (N1-AFU-86-17) Engineering Data Distribution and Control Records.

5. Department of the Air Force (N1-AFU-86-20). Air National Guard reenlistment bonus records.

6. Department of the Air Force (N1-AFU-86-21). Applications for ID cards and passes.

7. Department of the Navy, Headquarters U.S. Marine Corps (N1-127-86-2). Audio tapes of radio broadcasts.

8. Department of the Navy, Naval Data Automation Command (NC1-NU-84-2). A comprehensive schedule of all aeronautical and astronautical material records.

9. Department of State, Bureau of Consular Affairs, Visa Office (N1-84-86-1). Revision of disposition standards for certain categories of visa records maintained at Foreign Service posts.

10. Department of Transportation, Federal Aviation Administration (N1-237-86-2). Revision to standards for destruction of Federal Aid Airport Program/Airport Development Aid Program and Planning Grant Program records.

11. U.S. Postal Service, Finance Group (N1-28-86-1). Records used to develop volume forecasts and rate classifications.

12. Veterans Administration, (NC1-15-85-13). Plans and specifications relating to loans.

Dated: January 24, 1986.

Claudine Weiher,

Acting Archivist of the United States.

[FR Doc. 86-3113 Filed 2-11-86; 8:45 am]

BILLING CODE 7515-01-M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law (Pub. L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing this regular bi-weekly notice. Pub. L. 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately

effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This bi-weekly notice includes all amendments issued, or proposed to be issued, since the date of publication of the last bi-weekly notice which was published on January 29, 1986.

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.

By March 14, 1986, 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 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 of 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. 2.555, 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

presiding Atomic Safety and Licensing Board, that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., 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: April 17, 1985, as amended September 24, 1985.

Description of amendment request: This request supersedes the licensee's request dated March 20, 1984, which was noticed in the Federal Register on May 23, 1984 (49 FR 21825). The details of the ASME Boiler and Pressure Vessel Code Section XI Inservice Inspection (ISI) Program would be removed from the Technical Specifications by this amendment and placed in a controlled ISI Program document. The tables in the Technical Specifications listing snubbers on the Code Class 1, 2, and 3 systems would also be removed and placed in the controlled document. The revised Technical Specifications would then allow ISI changes for Code systems to be made without a subsequent Technical Specification change, which conforms to the approach use in the BWR Generic Technical Specifications in the area of ISI. However, this revision would not remove the requirements to perform ISI in accordance with Section XI and to test Code Class snubbers at required intervals.

As part of this amendment, the term "PNPS Procedure" would be substituted for references to the tables being removed. References to the 1974 Edition of the ASME Boiler and Pressure Vessel Code would be changed to the 1980 Edition, Winter 1980 addenda. Thus, the revised Technical Specifications will require conformance with changes which were made in the latter edition of the Code, which is incorporated by reference in 10 CFR 50.55a. The bases pages for these Technical Specifications would also be revised to be consistent with the foregoing changes.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of standards for determining whether license amendments are not likely to 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 hazards consideration is "(vii) A change to make a license conform to changes in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations."

The proposed amendment requires changes in the ISI program for the Pilgrim Station in keeping with changes in the ASME Boiler and Pressure Vessel Code, which is incorporated by reference in the Commission's regulations (10 CFR 50.55a). These changes in ISI requirements and shifting the tables of these requirements and snubbers from the Technical Specifications to a separate, controlled document are expected to have no effect on facility operations. This proposed amendment is, therefore, similar to example (vii) above and the staff proposes to determine that the application for this amendment does not involve a 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 Project Director: John A. Zwolinski.

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

Date of amendment request: December 23, 1985.

Description of amendment request: The proposed amendment would change the Technical Specifications administrative section relative to the licensee's Nuclear Safety Review and Audit Committee (NSRAC). The change would clarify the composition of NSRAC and its quorum requirements, identify specifically the types of safety evaluations to be reviewed by NSRAC, and delete a 14-day limit on the time allowed for preparation and distribution of the minutes of an NSRAC meeting.

Basis for proposed no significant hazards consideration determination: The proposed changes to the Technical Specifications are all administrative in nature and do not physically affect plant safety-related systems. Therefore, these changes would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a

margin of safety. Based on this finding, the staff has made an initial determination that the proposed amendment does not involve significant hazards considerations.

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 Project Director: John A. Zwolinski.

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

Date of amendment request:
December 23, 1985.

Description of amendment request:
The proposed amendment would change the Pilgrim Station Technical Specification by revising Note 1 to Table 3.1.1 and Note 1 to Table 3.2.A. These note changes would impose a time limit of 6 hours on keeping an instrument channel of the Reactor Protection System or the Primary Containment Isolation System out of service during testing and calibration. A time limit does not currently exist.

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 (April 6, 1983, 48 FR 14870). The examples of actions not likely to involve a significant hazards consideration include "(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."

The proposed change would add a restriction on the amount of time an instrument channel is deliberately made inoperable without placing the trip system in the tripped condition. Because this change is similar to example (ii), the staff has made a proposed determination that the proposed amendment would involve no significant hazard considerations.

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 Project Director: John A. Zwolinski.

Carolina Power & Light Company, Docket No. 50-324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of application for amendment:
December 20, 1985.

Description of amendment request:
The proposed amendment would change the Technical Specifications (TS) for the Brunswick Steam Electric Plant, Unit 2. The proposed revision to TS Tables 3.3.3-1, and 4.3.3-1 reflects modifications to be performed on the Brunswick 2 Automatic Depressurization Systems (ADS) during the Reload 6 outage commencing December 1, 1985. These modifications will remove the high pressure trip from the ADS logic sequence and add a manual inhibit switch, thereby, eliminating the need for manual ADS actuation to ensure core coverage. A similar request was granted for Brunswick 1 by Amendment No. 87 on July 30, 1985.

Currently, the ADS activates automatically upon coincident signals of low reactor vessel water level, high drywell pressure, and initiation of a low pressure emergency core cooling pump. Modifications to be performed on the ADS logic will remove the need for high drywell pressure indication for automatic initiation of the ADS. As a result, transients which do not directly produce a high drywell pressure signal will be encompassed by the operation of the ADS. A time delay of approximately 2 minutes after receipt of the signals allows the operator to reset the logic and prevent an unnecessary actuation.

The requested TS changes remove the ADS high drywell pressure instruments and add manual inhibit switches to the ADS in TS Tables 3.3.3-1, 3.3.3-2 and 4.3.3-1. A Boiling Water Reactor Owners Group (BWROG) study of alternatives to the present ADS actuation logic identified modifications to eliminate the need for manual actuation to ensure core coverage in the event of certain accident sequences. The proposed TS change is the second option outlined in the BWROG study and is one of the two options indicated to be acceptable by NRC letter dated June 3, 1983. The resulting reduction of logic devices will increase ADS reliability and will provide additional assurance of adequate core cooling by further automating reactor pressure vessel depressurization for certain system isolations and stuck open relief valve events, while satisfying design concerns associated with anticipated transients without scram.

Basis for proposed no significant hazards consideration determination:

Carolina Power & Light Company (CP&L) has determined that the requested amendment does not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because the modifications result in an enhancement of the ADS and does not affect performance of the intended safety function. As a result of these modifications, ADA operation will be extended to encompass accident and transient condition which do not produce a high drywell pressure signal.

2. Create the possibility of a new or different kind of accident than previously evaluated for the same stated in item 1.

3. Involve a significant reduction in the margin of safety because removing the need for manual actuation eliminates the possibility of operator error thereby ensuring core coverage. In addition, reduction of the number of logic devices increases ADS reliability.

Based on the above reasoning, CP&L has determined that the proposed changes involve no significant hazards consideration.

The staff has reviewed the CP&L determinations and finds that the amendment request meets the standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)), that is, the proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (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.

Based on the above discussion the Commission proposes to determine that the amendment does not involve a significant hazards consideration.

Local Public Document Room
location: Southport, Brunswick County Library, 109 W. Moore Street, Southport, North Carolina 28461.

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

NRC Project Director: Daniel Muller.

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

Date of amendment request: May 30, 1984, as supplemented December 19, 1985.

Description of amendment request: This submittal supplements the request for amendment dated May 30, 1984 which was noticed in the Federal Register on July 24, 1984 (49 FR 29906). The proposed Technical Specifications Revision would revise portions of Consolidated Edison's May 30, 1984 license amendment application to make the Specifications concerning Hydraulic Snubbers consistent with the Standard Technical Specifications. These changes were requested in a letter from NRC to Consolidated Edison dated October 11, 1985. The proposed changes would revise the specifications required testing of representative sample 10% of IP-2's safety related snubbers. The revision proposes to revise the specification to require that for each snubber found inoperable an additional 10% of that type of snubber shall be functionally tested. The revision also proposes to require retesting of failed snubbers and independent testing of snubbers with manufacturer or design defects.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards in 10 CFR 50.92 by providing certain examples (48 FR 14870). One of the examples (ii) of actions not likely to involve significant hazards considerations relates to a change that constitutes an additional limitation, restriction or control not presently included in the Technical Specification: for example, a more stringent surveillance requirement. The proposed revision to the Technical Specification concerning snubber testing is consistent with example (ii) in that the proposed change constitutes more stringent requirements.

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

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

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

NRC Project Directorate: Steven A. Varga.

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

Date of amendment request: December 12, 1985.

Description of amendment request: The amendment would modify the Technical Specifications (TS) to reflect the use of replacement control rods of the ASEA-ATOM (AA) plate design in

addition to the currently used Allis-Chalmers (A-C) tube sheath design. The AA control rods for the La Crosse Boiling Water Reactor (LACBWR) have been designed to closely match the reactivity worth of the original A-C control rods and to be mechanically compatible with all reactor components and control rod handling equipment. The amendment would also delete the requirement to go-gage control rods (i.e., gage the thickness to detect swelling), based on LACBWR and industry operating experience, and that Standard Technical Specifications for boiling water reactors do not contain a similar requirement.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The licensee has presented its determination of significant hazards considerations as follows:

10 CFR 50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis using the standards in 10 CFR 50.92, about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the following analysis has been performed.

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

Since the AA control rods are mechanically compatible in all respects with the LACBWR system and have essentially the same reactivity worth, their use will not involve a significant increase in the probability or consequences of an accident previously evaluated.

For negative reactivity insertion events (scram events) the slightly greater reactivity worth of the AA control rods is considered to be beneficial. Slightly more negative reactivity will be inserted faster than with the A-C control rods. The shutdown margin will also be slightly

greater with AA control rods than with the current A-C control rods.

For positive reactivity insertion events, i.e. inadvertent control rod withdrawals and control rod drop accident, the slightly greater worth of the AA control rod is also expected to have a minimal effect. The limiting anticipated transient for the LACBWR is the inadvertent control rod withdrawal at operating power. A recalculation of the limiting control rod withdrawal transient for the beginning of Fuel Cycle 9 using a conservative 3% (relative) greater control rod worth resulted in a Minimum Critical Power Ratio (MCPR) of 1.529 compared to a MCPR of 1.539 calculated for the A-C rods. The effect produced by the actual LACBWR AA control rod with a relative worth equal or slightly less than the A-C rods in the operating reactor would be completely negligible. In the LACBWR, the consequences of a control rod drop accident are greatest when the reactor is at power. The results of the probability study of the LACBWR control rod drop accident are not very sensitive to the specific worth of the control rods and the small differences between the AA and the A-C control rods would have an insignificant effect on the results.

The deletion of TS 4.2.4.10, on go-gaging control rods, will not increase the probability or consequences of an accident, since it is a surveillance requirement which LACBWR and industry experience has shown to be ineffective. The requirement to gage the control rods was originally based on the belief that the life of the rod would be limited by the pressure buildup in the B₄C tubes due to helium release from the irradiated B₄C and that the gage would detect tube swelling before tube failure. Experience has shown that absorber tubes can fail by intergranular stress corrosion cracking (IGSCC) and B₄C be lost before any swelling is detected by gaging.

The industry has concluded, after extensive research, that absorber tube failure by IGSCC and B₄C loss is more a function of B-10 depletion, and resultant B₄C swelling and change in physical characteristics, than a result of pressure buildup from helium release. Therefore, TS 4.2.4.10 is no longer necessary.

2. Operation of the LACBWR in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the AA control rods are mechanically compatible in all respects with the LACBWR system and have essentially the same reactivity worth as

the A-C control rods, their use will not create the possibility of a new or different kind of accident from any previously evaluated.

Elimination of the go-gaging surveillance requirement will not create the possibility of a new or different type of accident since industry experience has shown that swelling due to pressure buildup due to helium release is not the primary cause of absorber tube failure and that failures can occur prior to any swelling being detected by gaging.

3. *Operation of the LACBWR in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.*

The AA control rods for the LACBWR have been designed to closely match the reactivity worth of the original (A-C) control rods and to be mechanically compatible with all reactor components and control rod handling equipment. The AA control rod is slightly lighter (approximately 7 lbs.) than the A-C rod and therefore, scram times for the AA rod should be approximately the same as or slightly faster than for the A-C rod. Scram times for the AA control rods will be measured after installation in the reactor as required by current procedures and technical specifications.

The effect of the AA control rods on the margin of safety during an inadvertent control rod withdrawal transient, which is the limiting anticipated transient, for LACBWR, would be negligible. The limitations of other control rod related Technical Specifications such as 4.2.5.2 and 4.2.5.3 will still be conservatively met during operations with the AA control rods. Their improved design will significantly reduce the probability of IGSCC failure and resultant loss of control material (B,C) and therefore, will increase the margin of safety.

The deletion of the go-gaging requirement will not decrease the margin of safety, since industry experience has demonstrated that pressure buildup in B,C tubes due to helium release is not a dominant failure mode. This surveillance is not required by Standard Boiling Water Reactor (EWR) Technical Specifications. Control rod lifetime will be appropriately and prudently based on exposure histories, B-10 depletion and visual examinations as is currently the case for the rest of the U.S. BWR reactors. Therefore, the proposed amendment will not involve a significant reduction in the margin of safety.

As determined by the analysis above, this proposed amendment has no significant hazards consideration.

The staff has reviewed the licensee's significant hazards consideration determination and based upon this review, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.

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

Attorney for license: Roy P. Lessy, Jr.; O.S. Heistand; Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Project Director: John A. Zwolinski.

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

Date of amendment request: January 2, 1986.

Description of amendment request: The licensee proposed a technical specification amendment to permit a one time extension of the 12 month \pm 25% snubber visual inspection period (October 4, 1985 through April 4, 1986) to the fifth refueling outage, scheduled to begin in May 1986. The refueling outage is tentatively scheduled to begin within a short period of time beyond the currently specified snubber inspection period. Without the extension, plant shutdown would be needed to perform the visual inspections, since the snubbers inside containment are considered inaccessible during power operation due to the subatmospheric containment design. There has not been an outage of sufficient duration during the present fuel cycle to perform the required visual inspections.

Basis for proposed no significant hazards consideration determination: The proposed amendment would only extend slightly the period during which the snubbers will have to be inspected. It does not change the way the snubbers are to be inspected, nor does it reduce in any way inspection requirements. Consequently, the amendment would not result in a significant increase in the probability of occurrence of an accident or malfunction of equipment important to safety, would not create an accident of a different type, and would not significantly reduce the margin of safety of the plant. We, therefore, propose to characterize the proposed amendment as involving no significant hazards consideration.

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

Attorney for licensee: Gerald Charnoff, Esquire, Jay E. Silberg,

Esquire, Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, D.C. 20036.

NRC Project Director: Lester S. Rubenstein.

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

Date of amendment request: January 17, 1986.

Description of amendment request: The proposed amendments would revise the Technical Specifications by adding a provision to allow two plant operation with 3 out of 4 Essential Service Water (ESW) pumps operational and the two plants' ESW systems aligned in the cross-tie mode.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48 FR 14870). One of the examples 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 acceptable criteria with respect to the system or component specified in the Standard Review Plan. The proposed amendment is directly related to the example.

The design basis for the D.C. Cook Nuclear Plant is a loss of coolant accident (LOCA) in one unit and hot shutdown in the other and for this condition, only two of the four ESW pumps, one per Unit, are necessary. The three pumps in a cross-tie mode of operation, as proposed by this amendment, are capable of exceeding the design basis by supplying sufficient water for a LOCA in one unit and cooldown of the other unit. Therefore, the proposed amendment will allow operation which may reduce a margin of safety by not having the fourth pump operable but having three pumps in a cross-time mode is clearly within the acceptable criteria for the Essential Service Water system at the D.C. Cook Nuclear Plant. Therefore, the Commission proposes that the changes do not involve a significant hazard consideration.

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

Attorney for licensee: Gerald Charnoff, Esquire, Shaw, Pittman, Potts

and Trowbridge, 1800 M Street, N.W., Washington, D.C. 20036.

NRC Project Director: B.J. Youngblood.

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

Date of amendment request:
December 30, 1985.

Description of amendment request:
The proposed amendment would revise the Duane Arnold Energy Center (DAEC) Technical Specifications reflecting the DAEC conformance with 10 CFR 50, Appendix R, Section III.G related to fire protection program. Specifically, changes in the Technical Specifications relate to (a) Remote Shutdown Panels, (b) Automatic Fire Suppression System, (c) Heat Detectors and Ionization Smoke Detectors, and (d) Administrative changes required for the fire protection program. The proposed changes are described as follows:

(a) Remote Shutdown Panels. The Remote Shutdown Panels (RSPs) are required to achieve and maintain cold shutdown of the DAEC nuclear reactor in the unlikely event that the main Control Room becomes uninhabitable or is damaged by fire. The proposed Technical Specification change provides for periodic inspection and testing of the panels to verify operability. The inspection and operability requirements are consistent with those currently required for existing safe shutdown instrumentation.

The proposed inspection and testing requirements for the RSPs replace the existing requirements for the existing Emergency Shutdown Control Panel (ESCP) since the RSP system now incorporates the function of the ESCP. Incorporating the ESCP into the RSP system makes the ESCP one of the local control panels in the RSP system. Likewise, the proposed bases change the wording of paragraphs 3.10.B and 4.10.B to reflect the RSP system rather than the ESCP.

(b) *Automatic Fire Suppression Systems.* The Automatic Fire Suppression Systems are required to protect safety-related systems required for safe plant shutdown and must be operable whenever safe shutdown equipment in the protected area is required to be operable. The proposed Technical Specification change provides for periodic inspection and testing to verify operability of these new fire suppression systems.

(c) *Heat Detectors and Ionization Smoke Detectors.* These fire detection systems are required to protect safety-related systems when the safe shutdown equipment in the protected area is

required to be operable. The proposed Technical Specification change provides for periodic testing and inspection to verify operability of the fire detection systems. The bases on page 3.13-10 have been revised to incorporate the new fire detection instrumentation and provide compatibility with Table 3.13-1.

(d) *Administrative Changes.* The proposed Technical Specification change corrects three typographical errors which consist of an unneeded period on page iii, a misspelling of the word "detection" on page 3.13.1, and an error in the fire pump discharge nozzle head value. Correction of the discharge pressure is needed to be in conformance with the correct value found in the Final Safety Analysis Report (FSAR) and the manufacturer's pump curve. One page 3.13-10 and in Table 3.13-1, the "Control Auxiliary Panel Room" has been renamed the "Control Room Back Panel Area" to conform to terminology used by plant personnel. The existing "Zones" in Table 3.13-2 have been changed to "Fire Detection Zones" to distinguish them from the Fire Zones listed in the DAEC Fire Hazards Analysis (FHA).

Basis for proposed no significant hazards consideration determination:
The Commission has provided standards (10 CFR 50.92(c)) for determining whether a significant hazards consideration exists. 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.

We have reviewed the license's request and find that the proposed amendment:

(1) Does not involve a significant increase in the probability or consequences of an accident previously evaluated, because:

(a) The Remote Shutdown Panels are required to achieve and maintain cold shutdown of the reactor in the unlikely event that the main Control Room becomes uninhabitable or is damaged by fire and the proposed Technical Specification change provides for periodic inspection and testing to verify operability of the RSP, and constitutes an additional restriction in the Technical Specifications.

(b) The Automatic Fire Suppression Systems are required to protect certain safety-related systems, in case of fire,

for safe plant shutdown, and the Technical Specification changes consisting of periodic testing and inspection to verify operability of the new fire protection systems, constitute additional restrictions in the Technical Specifications.

(c) Heat Detectors and Ionizing Smoke Detectors will be tested in the same manner as the existing detectors, and

(d) Administrative changes consist of an unneeded period on page iii, a misspelling of the word "detection" on page 3.13-1, and typographic correction of the fire pump nozzle discharge pressure value. In addition, changing the name of the "Control Auxiliary Panel Room" to the "Control Room Back Panel Area" conforms to terminology used by plant personnel and will eliminate a possible source of confusion, and changing the existing "Zones" in Table 3.13-1 to "Fire Detection Zones" clarifies the distinction between these detector zones and the "Fire Zones" described in the Fire Hazards Analysis (FHA).

(2) Does not create a possibility of a new or different kind of accident because:

(a) Remote Shutdown Panels (RSPs) utilize plant systems which have already been evaluated by the staff and the RSPs are inspected and tested in similar manner as the Control Room instrumentation,

(b) Automatic Fire Suppression Systems is an expansion of the existing Automatic Fire Suppression System, which has been evaluated by the staff,

(c) Fire detection system Technical Specification change provides for periodic testing and inspection to verify operability of these fire detection systems in the same manner as the existing detectors which have been evaluated by the staff, and

(d) Administrative changes described previously do not alter the meaning of the existing Technical Specifications.

(3) Does not involve a significant reduction in a margin of safety because:

(a) The inspection and testing of the Remote Shutdown Panels (RSPs) should increase the margin of plant safety since the RSPs provide increased capability to place the reactor in cold shutdown in the unlikely event that Control Room becomes uninhabitable or is damaged by fire. Furthermore, the staff has reviewed this method of alternate shutdown for BWRs and has issued a Safety Evaluation Report, dated January 6, 1983, approving its use at the DAEC.

(b) The inspection and testing of Automatic Fire Suppression Systems equipment does not reduce the margin of safety.

(c) The testing of the fire detection system will improve the margin of safety for protection of the safety-related equipment and components which are part of the safe shutdown systems, and

(d) Administrative changes described previously do not affect margins of safety.

Therefore, the staff has made a proposed determination that the application involved no significant hazards consideration.

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

Attorney for licensee: Jack Newman, Esquire, Harold F. Reis, Esquire, Newman and Holtzinger, 1025 Connecticut Avenue, N.W., Washington, D.C. 20036.

NRC Project Director: Daniel R. Muller.

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

Date of amendment request: December 6, 1985, as supplemented January 13, 1986.

Description of amendment request: The amendment would modify the Technical Specifications (TS) to reflect the addition of Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the General Electric fuel bundle, type P8DRB299. These limits were calculated by using the same approved General Electric methods used for the present fuel type P8DNE277. This proposed amendment would allow the use of P8DRB299 in the upcoming and other future fuel cycles since appropriate MAPLHGR limits would be provided for this fuel type.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The licensee has presented its determination of significant hazards considerations as follows:

10 CFR 50.91 requires that at the time a licensee requests an amendment, it

must provide to the Commission its analysis using the standards in 10 CFR 50.92, about the issue of no significant hazards consideration. Therefore, in accordance with 10 CFR 50.91 and 10 CFR 50.92, the following analysis has been performed.

1. *The proposed amendment in accordance with the operation of Nine Mile Point Unit 1 will not involve a significant increase in the probability or consequences of an accident previously evaluated.* The methods used to analyze the Loss of Coolant Accident response of the P8DRB299 fuel conform to Appendix K requirements and are identical to those previously used. Results for the type P8DRB299 fuel analysis are included as Figure 3.1.7(f). The peak cladding temperature and maximum oxidation fraction limits are approximately the same as for previous fuel types. Therefore, the proposed amendment will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. *The proposed amendment in accordance with the operation of Nine Mile Point Unit 1 will not create the possibility of a new or different kind of accident from any accident previously evaluated.*

Results for the type P8DRB299 fuel analysis demonstrate that the Loss of Coolant Accident response is approximately the same as for the fuel currently used. The peak cladding temperature and maximum oxidation fraction limits are insignificantly different, and therefore, (the P8DRB299 fuel type) constitute a one-for-one replacement with the currently used fuel. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed amendment in accordance with the operation of Nine Mile Point Unit 1 will not involve a significant reduction in the margin of safety.*

An analysis of the Loss of Coolant Accident response of proposed fuel bundle type P8DRB299 has been completed in accordance with methods previously used. The results of the analysis show that the peak cladding temperature and the maximum oxidation fraction limits are within the limits set by Appendix K and are approximately the same as those previously accepted. Therefore, the proposed amendment in accordance with the operation of Nine Mile Point Unit 1 will not involve a significant reduction in the margin of safety.

As determined by the analysis above, this proposed amendment has no significant hazards consideration.

The staff has reviewed the licensee's significant hazards consideration determination and based upon this review, the staff has made a proposed determination that the application for amendment involves no significant hazards consideration.

Local Public Document Room
location: State University College at Oswego, Penfield Library — Documents, Oswego, New York 13126.

Attorney for licensee: Troy B. Conner, Jr., Esquire, Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue, N.W., Washington, D.C. 20006.

NRC Project Director: John A. Zwolinski.

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

Date of amendments request: October 16, 1985.

Description of amendments requests: The proposed amendments request would revise the technical specifications regarding the method of calibration of the Analog Rod Position Indication Systems by providing more realistic operational requirements consistent with the safety requirements of the system. Specifically, the Unit 1 Technical Specifications would be modified as follows:

Replace 3.1.3.1 through 4.1.3.2 with revised Tech Specs 3.1.3.1 through 4.1.3.2.

Tech Specs 3.1.3.3 through 4.1.3.3 remain unchanged.

Replace 3.1.3.4 through 4.1.3.5 with revised Tech Specs 3.1.3.4 through 4.1.3.5.

Tech Spec Figures 3.1-1 and 3.1-2 remain unchanged.

Replace $\frac{1}{4}$.1.3 with revised Bases $\frac{1}{4}$.1.3

The Unit 2 Technical Specifications would be modified as follows:

Replace 3.1.3.1 through 4.1.3.2.2 with revised Tech Specs 3.1.3.1 through 4.1.3.2.2.

Tech Spec 3.1.3.3 through 4.1.3.3 remain unchanged.

Replace 3.1.3.4 through 4.1.3.5 remain revised Tech Spec 3.1.3.4 through 4.1.3.5.

Tech Specs Figures 3.1-1 and 3.1-2 remain unchanged.

Replace Bases $\frac{1}{4}$.1.3 with revised Bases $\frac{1}{4}$.1.3.

Basis for proposed no significant hazards consideration determination: Shutdown Banks and Control Banks A and B positions need to be known

accurately in a very limited range, near the top and bottom of the core. Accurate knowledge of these bank positions permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position.

Recognizing that the Analog Rod Position Indication (ARPI) is very temperature sensitive, immediate verification of position after rod movement is shifted (in the revised Technical Specifications) from the ARPI to the group step counters with subsequent verification by the ARPI after temperature equilibration. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the ARPI (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits as assumed in the accident analyses.

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 an accident previously evaluated; or (3) Involve a significant reduction in a margin of safety. The licensee has determined and the staff agrees that the requested amendments per 10 CFR 50.92 do not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated for the Salem Units, since there are no changes to conditions assumed in the accident analyses; (2) Create the possibility of a new or different kind of accident from any accident previously evaluated for the Salem Units, since no plant modifications resulted from change; or (3) Involve a significant reduction in a margin of safety, since there are no changes to conditions assumed in the accident analyses.

Accordingly, the Commission proposed to determine the proposed changes to the Technical Specifications involve no significant hazards considerations.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, N.W., Washington, D.C. 20006.

NRC Project Directorate: Steven A. Varga.

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

Date of amendments request: October 17, 1985.

Description of amendments request: The amendments request would change Technical Specifications Section 4, Table 4.3-1 for Salem Units Nos. 1 and 2 by adding surveillance requirements for the Reactor Trip Circuitry not presently included in the Technical Specifications. These additional requirements are responsive to NRC conclusions identified in Items 10 and 13 of the staff's Safety Evaluation dated June 25, 1984, that responded to the Public Service Electric and Gas Company's submittal regarding Item 4.3 of Generic Letter 83-28, "Reactor Trip Breaker Automatic Shunt Trip."

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). The examples of actions which involve no significant hazards consideration include a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications. The proposed changes add surveillance requirements for the Reactor Trip Circuitry not presently included in the technical specifications. Therefore, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Salem Free Library, 122 West Broadway, Salem, New Jersey 08079.

Attorney for licensee: Conner and Wetterhann, Suite 1050, 1747 Pennsylvania Avenue, N.W., Washington, D.C. 20006.

NRC Project Directorate: Steven A. Varga.

Southern California Edison Company et al., Docket No. 50-206, San Onofre Nuclear Generating Station, Unit No. 1 (SONGS 1), San Diego County, California

Date of amendment request: November 21, 1985.

Description of amendment request: The proposed change would delete the

interim surveillance testing requirement established on November 5, 1981 which requires that Safety Injection System (SIS) valves be tested once every 92 days. Deletion of the interim testing requirement would allow valve operability testing to be performed in accordance with existing Technical Specification 4.2.1 at intervals no longer than normal plant refueling intervals, which are typically 18 months in duration. The proposed change would also modify the surveillance testing procedure to require that valve actuation be accomplished in three to five seconds in order for the test to be successful.

Basis for proposed no significant hazards consideration determination: The interim SIS valve testing frequency was established by the NRC on November 5, 1981 in order to evaluate the effectiveness of a design modification made to the SIS valves to increase their reliability. The interim program required that a long-term testing program based upon the results of the interim testing be established at the next refueling outage. Since SONGS 1 did not operate from February 1982 until November 1984, the next refueling outage began in November 1985. During the past fuel cycle, the SIS valves were tested six times and each time the actuator force required to open the valves was well within the capacity of the valve actuators. Thus, the licensee has concluded that the design modification has been verified and the valve testing frequency may be reduced from every 92 days to once each refueling shutdown. The valve actuation time limitation is added in order to ensure that the valves actuate within the time constraints approved by the staff's November 5, 1981 Safety Evaluation.

Based upon the above discussion, the staff has concluded that the operation of the facility in accordance with the proposed license 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 previously evaluated; or (3) involve a significant reduction in the margin of safety.

Accordingly, the Commission's staff proposes to determine that the proposed changes to the Technical Specifications involve no significant hazards considerations.

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

Attorney for licensee: Charles R. Kocher, Assistant General Counsel,

James Beoletto, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

NRC Project Director: George E. Lear.

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

Date of Amendment Request: March 20, 1984, April 27, 1984 and October 10, 1985 (Reference PCN-8).

Description of Amendment Request: The proposed change would revise Technical Specification (TS) 3/4.6.1, "Containment Integrity" and 3/4.6.3, "Containment Isolation Valves." TS 3/4.6.1 requires that containment integrity be maintained when the plant is in the hot shutdown, hot standby, startup and power operation modes (Modes 1-4) and specifies surveillance requirements to verify containment integrity and actions to be taken when the requirements are not met. These requirements ensure that offsite doses resulting from postulated accidents are bounded by the accident analyses.

TS 3/4.6.3 requires that the containment isolation valves listed in Table 3.6-1 be operable when the plant is in Modes 1-4 and specifies surveillance requirements to verify operability and actions to be taken when the operability requirements are not met. The operability of containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a postulated accident. PCN-8 makes several revisions to TS 3/4.6.1 and 3/4.6.3 and the associated Bases. These changes are summarized as follows:

1. Reorganization of Table 3.6-1.
2. Removal of the Purge Isolation Valves from Section A.
3. Removal of Secondary System from Table 3.6-1.
4. Removal of Shutdown Cooling Relief Valves from Table 3.6-1.
5. Determination of "isolation time" from the Limiting Condition for Operation (LCO).
6. Restriction of Applicability of the existing action requirements to Sections A, B and C of Table 3.6-1.
7. Clarification of "penetration."
8. Revision of Surveillance Requirements.
9. Addition of a TS 3.0.4 Exception.
10. Correction of Typographical Errors.
11. Definition of "secured" and deletion of "deactivated."

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 (i) relates to 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. Example (vi) relates to a change which may either result in some increase in the probability or consequences of a previously analyzed accident or may, in some way, reduce a safety margin but where the results of the change are currently within all acceptance criteria with respect to a system or component provided in the Standard Review Plan (SRP); for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. Each of the changes identified above is similar to one of these examples, therefore, it is proposed that the proposed changes do not involve Significant Hazards Considerations. A more detailed description of each of the proposed changes and why each is similar to one of the examples is provided below:

(1) Reorganization of Table 3.6-1. The existing Table 3.6-1 consists of four sections: (A) Containment Isolation; (B) Containment Purge; (C) Manual; and (D) Other. The containment isolation valves are categorized into Table 3.6-1. The proposed change redefines the four sections of Table 3.6-1 and reassigns the containment isolation valves remaining in Table 3.6-1 accordingly. The proposed Section A is titled "Automatic Containment Isolation." Included in Section A are automatic containment isolation valves which are actuated by a containment isolation actuation signal (CIAS). Also included in Section A are check valves located inside containment which are considered to be automatic containment isolation valves from the standpoint of the 10 CFR 50 Appendix A general design criteria. Section B of the proposed table remains titled "Containment Purge" and includes the containment purge isolation valves. The proposed Section C remains titled "Manual" isolation valves and includes those manual containment isolation valves which are assumed to be closed post-accident and can be opened intermittently during normal operation under administrative control. Section D remains titled "Other" and includes those valves whose post-accident position may be open and whose

operability requirements are defined by technical specifications other than 3.6-3, "Containment Isolation Valves." As a result of the proposed redefinition of the containment isolation valve table section described above, many valves have been moved from one category to another. The valves remaining in Sections A, B and C of the table continue to be subject to the current action and surveillance requirements. Valves which are relocated to Section D of the table have new action and surveillance requirements which reference the technical specifications governing their primary function which is other than containment isolation. The reorganization moves the following valves, which are currently listed in Section D, to Section A: demineralized water check valve, 3"-238-C-875, fire protection check valve 4"-061-C-681, quench tank makeup valve, 2"-573-C-611, service air supply line check valve 2"-017-C-627, instrument air supply line check valve 1½"-016-C-617, LP nitrogen check valve ¾"-002-C-611, component cooling water inlet isolation valve HV 6223, component cooling water outlet Isolation Valve 2"-129-A-544, and nitrogen supply to safety injection tank check valve 2"-108-C-627. No response time is included for the check valves noted above. Response times of 40 seconds are included for HV 6233 and HV 6236.

The proposed reorganization of Table 3.6-1 discussed above is similar to Example (i) of 48 FR 14870. The valves remaining in Sections A, B and C of Table 3.6-1, existing actions and surveillance requirement items as discussed below, continue to apply to these valves. The valves relocated or remaining in Section D of the table are subject to TS surveillance and action requirements for the systems in which they are included. The redefinition of Section D includes those valves whose normal safe post-accident safe position is open, the action requirements for the systems which include these valves would have these valves maintained opened if inoperable. However, the existing TS 3.6.3 action requirements, which currently apply to these valves, would require these valves to be closed if inoperable. The proposed change eliminates the applicability of the existing TS 3.6.3 action requirements to the valves included in Section D of Table 3.6-1. The proposed change would apply the action requirements corresponding to their primary system function which would maintain the valves in safe position, whereas existing requirements would force closure of an inoperable valve rendering the system

inoperable. Because this change resolves an inconsistency in the technical specifications, it is similar to Example (i) of 48 FR 14870.

(2) Removal of the Purge Isolation Valves from Section A. Section A currently includes the containment mini-purge isolation valves HV 9821., HV 9823, HV 9824 and HV 9825. These valves are deleted from Section A of the table because they are also included in Section B of the table and their operability is also covered by TS 3/4.6.1.7, "Containment Ventilation System" and TS 3/4.3.2, "Engineered Safety Features Actuation System Instrumentation." This change eliminates unnecessary repetition of requirements in the technical specification and is therefore editorial and similar to Example (i) of 48 FR 14870.

(3) Removal of Secondary System from Table 3.6-1. Table 3.6-1 currently includes all main steam system related isolation valves that are associated with containment penetrations. These valves close on a main steam isolation signal to limit RCS cool down during postulated main steam line break events. In addition, certain main steam system valves, such as the main steam isolation valve and backup feedwater isolation valve, receive a containment isolation actuation signal on containment pressure high and close on a main steam line break inside containment. This mitigates the consequences of main steam system piping ruptures inside containment and prevents containment overpressurization due to such events. Response times for all main steam system related valves that receive an MSIS or CIAS are included in Table 3.3-5, "ESFAS Instrumentation Response Time" which is part of Technical Specification 3/4.3.2. In addition, Technical Specification 3/4.7.1.5 specifically addresses operability of the main steam isolation valves. The main steam relief valves are also deleted from Table 3.6-1 in that their operability requirements are specified in Technical Specification 3/4.7.1.1, "Main Steam Relief Valves."

The result of this change will mean that the containment isolation valve action requirements no longer would apply to main steam system valves. However, Technical Specification 3/4.3.2 will require that those valves which receive a MSIS and/or a CIAS are operable with specified response times. In addition, more specific requirements exist as noted for the MSIV's and main steam relief valves. Because the containment isolation valves will not longer apply directly to main steam

system related valves, this change constitutes a relaxation in existing requirements. In this case, acceptance criteria related to the proposed change is found in SRP Section 6.2.4 "Containment Isolation System" and SRP Section 16, "Technical Specifications." SRP Section 6.2.4 requires that the reviewer determine that technical specifications for containment isolation valves are adequate. SRP Section 16.0 considers technical specifications consistent with the standard technical specifications to be acceptable. In this case, the standard technical specifications do not specifically address whether the main steam related isolation valves be included in Table 3.6-1. There is considerable variance between recently licensed but similar plants. For example, some units licensed prior to SONGS 2 and 3 and at least one unit licensed following SONGS 2 and 3 do not have the main steam system related isolation valves included in Table 3.6-1. The main steam system related isolation valves do not serve a containment isolation function in the same sense as isolation valves on systems which communicate directly with the reactor coolant system or containment atmosphere. The primary function of the main steam related valves is to limit RCS overcooling and overpressurization of the containment barrier in main steam line break events. The calculational assumptions for these events relate to the response times of these valves. One means of ensuring that response times are adequately covered by the technical specifications is to include the main steam isolation valves in Table 3.6-1. However, as noted above, response times for these valves are included in other technical specifications. Also, the critical requirements for main steam system related isolation valves are adequately covered by the technical specifications without their inclusion in Table 3.6-1. Therefore, the proposed change ensures that operability requirements for the main steam system related isolation valves are adequately covered by technical specifications and in a manner consistent with the standard technical specification. Thus, the acceptance criteria of SRP Sections 6.2.4 and 16.0 are satisfied and the proposed change is similar to Example (vi) of 48 FR 14870.

(4) Removal of Shutdown Cooling Relief Valve from Table 3.6-1. The shutdown cooling system relief valve PSV9349 is currently included in Section D of Table 3.6-1. Operability requirements for the shutdown cooling system relief valve are defined in TS 3/

4.4.8.3, "Overpressure Protection Systems." The proposed change deletes PSV9349 from Table 3.6-1 since the operability requirements are included in Technical Specification 3/4.4.8.3. This change eliminates unnecessary redundancy in the technical specifications and is editorial and similar to Example (i) of 48 FR 14870.

(5) Deletion of "isolation time" from the Limiting Condition for Operation (LCO) LCO 3.6.3 requires that the containment isolation valves specified in Table 3.6-1 be operable with isolation times as shown in Table 3.6-1. The proposed change will require that the containment isolation valves specified in Table 3.6-1 be operable and deletes the reference to isolation times. The operability of containment isolation valves is verified by performance of the surveillance requirements. Proposed surveillance requirement 4.6.3.3 requires that the isolation time for each automatic valve listed in Sections A and B of Table 3.6-1 be determined within its limit when tested in accordance with the Inservice Inspection (ISI) Program. Because performance of the surveillance requirements is a condition of operability, it is redundant to include the response time requirement in the LCO. Because this change eliminates unnecessary redundancy, it is editorial and is similar to Example (i) of 48 FR 14870.

(6) Restriction of Applicability of Existing Action Requirements to Sections A, B and C of Table 3.6-1. The existing action requirements, which apply to all containment isolation valves listed in Table 3.6-1, require that an operable containment isolation valve be maintained in any penetration that is open and that inoperable valves be either restored to operable status, the penetration be isolated, or that the plant be in hot standby and cold shutdown within specified periods of time.

The proposed change would restrict the applicability of the action requirements to Section A, B and C of the proposed table. As noted above, Section D of the table includes those valves whose safe post-accident position may be open and whose operability requirements and actions are included in other specifications addressing the primary functions of the systems in which they are included. As a result, isolation of a penetration in the event of inoperability of one of the valves included in Section D may conflict with the primary function technical specification action requirements and is likely not the safest position for the valve. The proposed change references the action

requirements for the LOC's pertaining to the valves or systems in which the valves in Section D are installed. This change eliminates the existing conflict in the existing technical specification and, therefore, is similar to Example (i) of 48 FR 14870.

(7) Clarification of Penetration. The action requirements refer to the affected penetration when a containment isolation valve is inoperable. Many penetrations branch prior to containment isolation valves; thus each penetration may have one or more flow paths into containment associated with it. It is inappropriate to isolate all branches of such a penetration when a containment isolation valve is inoperable in only one branch. The existing word "penetration" may be misinterpreted to require that all branches be isolated. The proposed change resolves this potential misinterpretation by adding a note which defines a "penetration" as any flow path from the atmosphere or a piping system inside of containment to the atmosphere or a piping system outside of containment. Each flow path is considered as a separate penetration. Because the proposed change clarifies the existing specification and eliminates the possibility of misinterpretation, it is editorial and similar to Example (i) of 48 FR 14870.

(8) Revision of Surveillance Requirements. Currently there are three surveillance requirements for containment isolation valves. TS 4.6.3.1 requires that prior to returning an inoperable valve to service that it be demonstrated open by performing a cycling test and verification of isolation time. TS 4.6.3.2 requires that each isolation valve be demonstrated operable by verifying action on the appropriate ESF actuation signal. TS 4.6.3.3 requires that the isolation time of each power operated or automatic valve in Table 3.6-1 be determined within its limit when tested in accordance with the Inservice Inspection Program. These surveillance requirements are inadequate in that they fail to address the operability requirements for all valves that are not power operated or automatic isolation valves. The proposed change would institute surveillance requirements specifically addressing each type of isolation valve included in Table 3.6-1. The proposed TS 4.6.3.1 would require the isolation valve specified in Sections A and B of Table 3.6-1 (automatic containment isolation valves and containment purge isolation valves) to be demonstrated operable prior to returning an inoperable valve to service by

performance of testing in accordance with the Inservice Inspection Program. This includes verification of isolation time where applicable. Additionally, valves secured in their actuated position are considered operable pursuant to this specification. The proposed TS 4.6.3.2 would require each isolation valve specified in Sections A and B of Table 3.6-1 except check valves, to be demonstrated operable by verifying that on a ESFAS test signal (CIAS, SIAS or CPIS as appropriate) each isolation valve actuates to its isolation position. This requirement corresponds to existing Specification 4.6.3.2. The proposed TS 4.6.3.3 requires that the isolation time of the valves (except check valves) included in Sections A and B, would be response time tested in accordance with the ISI Program. This surveillance requirement corresponds to the existing TS 4.6.3.3. A new surveillance requirement 4.6.3.4 is proposed to address operability of manual isolation valves specified in Section C of Table 3.6-1. This new surveillance requirement references existing surveillance requirements 4.6.1.1.A which requires verification of position on a routine basis and 4.6.1.2.D which requires leak rate testing in accordance with 10 CFR 50 Appendix J. The proposed TS 4.6.3.5 addresses surveillance requirements for the valves included in Section D. These valves shall be demonstrated operable in accordance with the ISI Program and surveillance requirements associated with those LCO's pertaining to each valve or the system in which it is installed. Valves secured in the ESFAS actuated position (i.e., safe post-accident position) are considered to be operable pursuant to this surveillance requirement. Again, TS 4.6.3.5 introduces no new requirements since the ISI Program is required by Specification 4.0.5 and the LCO's pertaining to the primary function of valves included in Section D already exist. The proposed surveillance requirements would consider valves secured in their safe positions to be operable. This is consistent with the definition of operable in that these valves are capable and, in fact, are performing their specified functions. Since the proposed surveillance requirements do not modify existing surveillance requirements, and restate other existing surveillance requirements for Sections C and D valves, the proposed change is editorial and is similar to Example (i) of 48 FR 14870.

(9) Addition of an Exception to TS 3.0.4. TS 3.0.4 prevents the plant from being taken to a higher operational

mode while relying on the provisions of an action statement. The proposed change would add an exception to TS 3.0.4 in TS 3.6.3. As noted above, a valve which is secured in its safe position is performing its specified function and, therefore, is operable in accordance with operability definition. Thus, complying with the action requirements which require valves to be secured in their safe position, constitutes meeting the operability requirements. Therefore, once the action requirements are satisfied, the action is exited since the operability requirement is met. Thus, complies with the action statement does not restrict upward mode changes. Consistent with this, the TS 3.0.4 exception is added. Because the proposed change improves consistency within the technical specifications, the proposed change is similar to Example (i) of 48 FR 14870.

(10) Correction of Typographical Errors.

In Section C of Table 3.6-1 for Unit 2 only, the designation of Penetration 10C is changed to Penetration 10B to correct an existing error. For both units, the designations of hot leg injection isolation valves 3"-157-A-551 and 3"-158-A-551 are corrected to 3"-157-A-550 and 3"-158-A-550, respectively. These changes correct an existing error and are therefore similar to Example (i) of 48 FR 14870.

(11) Definition of Secure and Deletion of Deactivated.

Technical Specification 3/4.6.3 Action 1B currently requires that an affected penetration be isolated by the use of at least one deactivated automatic valve secured in the isolation position. Surveillance requirement 4.6.1.1.A requires that all penetrations not capable of being closed by automatic containment isolation valves and require to be closed during accident condition are closed by valves, blind flanges, or deactivated automatic valves secured in their positions. The proposed change would add notes to define secured as being locked, secured or otherwise prevented from unintentional operation. This definition of secured would be used in place of the word deactivated which is deleted from these specifications by the proposed change. Deletion of deactivated and institution of the definition of secured will prevent misinterpretations of the term deactivated.

The word deactivated can be interpreted to mean that an automatic valve closed be closed and deenergized for actuation with its main circuit breaker locked open. While this effectively prevents the valve from

unintentional operation, it also, in many cases, deenergizes the position indication circuits. Verification of the position of valves inside containment, for example, would not be possible during operation in this condition. There are several ways of preventing valves from spurious or unintentional operation short of deactivating them by racking out the breaker. The definition of the word secured will provide the necessary facility to maintain position indication which taking measures to secure the valves and avoid misinterpretation of the existing word deactivated. Because this change merely clarifies existing requirements, i.e., valves will still be required to be prevented from unintentional operation by some mechanism, the proposed change is editorial in nature and is similar to Example (i) 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 Project Director: George W. Knighton.

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 considerations.

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.

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

Date of amendment request:

December 12, 1985.

Description of amendment request:

The proposed amendment would increase the maximum average exposure of any fuel assembly not on the

periphery of the core from 16,800 MWD/MTU to 18,000 MWD/MTU.

Date of publication of individual notice in Federal Register: January 21, 1986 (51 FR 2776).

Expiration date of individual notice: February 20, 1986.

Local Public Document Room

Location: La Crosse Public Library, 800 Main Street, La Crosse, Wisconsin 54601.

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

Date of amendment request:

November 27 and December 10, 1985.

Brief description of amendment:

Technical Specification changes relating to the allowable range for the moderator temperature coefficient.

Date of publication of individual notice in Federal Register: December 27,

1985 (50 FR 53031).

Expiration date of individual notice:

January 27, 1986.

Local Public Document Room

Location: San Clemente Library, Avenida Del Mar, San Clemente, California 92672.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

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

Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the Federal Register as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these

amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Alabama Power Company, Docket Nos. 50-348 and 50-364, Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2, Houston County, Alabama

Date of application for amendments: September 19, 1985.

Brief description of amendments: The Administrative Controls section of the Technical Specifications were revised to reflect retitles of on-site and off-site licensee management. Other minor reorganizational changes were made for plant maintenance activities, and computer services and operations functions.

Date of issuance: January 27, 1986.

Effective date: January 27, 1986.

Amendment Nos.: 60 and 51.

Facilities Operating License Nos. NPF-2 and NPF-8. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register. November 6, 1985 (50 FR 46209).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1986.

No significant hazards consideration comments were received.

Local Public Document Room

Location: George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303

Arizona Public Service Company, et al. Docket No. STN 50-528, Palo Verde Nuclear Generating Station, Unit 1, Maricopa County, Arizona

Date of application for amendment: October 16, 1985.

Brief description of amendment: The amendment extends the deadline for environmental qualification of electrical

equipment from November 30, 1985 to March 30, 1986.

Date of issuance: January 29, 1986.

Effective date: November 18, 1985.

Amendment No.: 4

Facility Operating License No. NPF-41. Amendment revised the license.

Date of initial notice in Federal Register. December 18, 1985 (50 FR 51619).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 1986.

No significant hazards consideration comments were received.

Local Public Document Room

Location: Phoenix Public Library, Business, Science and Technology Department, 12 East McDowell Road, Phoenix, Arizona 85004.

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

Date of application for amendments: November 19, 1985.

Brief description of amendments: The amendments permit a one-time exception to the technical specifications for the purpose of making environmental qualification modifications to the hydrogen recombiner system.

Date of issuance: January 27, 1986.

Effective date: January 27, 1986.

Amendment Nos.: 5 (Palo Verde Unit 1) and 1 (Palo Verde Unit 2).

Facility Operating License Nos. NPF-41 and NPF-46. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register. December 18, 1985 (50 FR 51619).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 27, 1986.

No significant hazards consideration comments were received.

Local Public Document Room

Location: Phoenix Public Library, Business, Science and Technology Department, 12 East McDowell Road, Phoenix, Arizona 85004.

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

Date of application for amendments: June 28, 1985.

Brief description of amendments:

These amendments would modify Section 3.22, 4.22 and 6.5.B. of the Technical Specifications, relating to mechanical and hydraulic snubbers, to

conform with Standardized Technical Specifications.

Date of issuance: January 22, 1986.

Effective date: January 22, 1986.

Amendment Nos.: 93 and 83.

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

Date of initial notice in Federal Register. July 31, 1985 (50 FR 31068).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 22, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room

Location: Zion Benton Library District, 2600 Emmaus Avenue, Zion, Illinois 60099.

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 7, 1985.

Brief description of amendment:

Revises the Technical Specifications to change the limiting conditions for operation (LCO's) for containment cooling and Iodine Removal Systems and associated containment isolation provisions.

The amendment also contains editorial changes for consistency with the language used in other of the Indian Point 2 Technical Specifications.

Date of issuance: January 27, 1986.

Effective date: Immediately to be implemented within 30 days.

Amendment No.: 108.

Facilities Operating License No.

DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 14, 1985 (50 FR 32791).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room

Location: White Plains Public Library, 100 Martine Avenue, White Plains, New York, 10610.

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

Date of application for amendment: August 28, 1985.

Brief description of amendment: The amendment changes the Technical Specifications to permit an exception to the experience requirements for six identified candidates for senior reactor operator licenses.

Date of issuance: January 24, 1986.

Effective date: January 24, 1986.

Amendment No.: 2.

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

Date of initial notice in Federal Register: November 6, 1985 (50 FR 46212). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room

Location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

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

Date of application for amendment: November 17, 1981.

Brief description of amendment: This amendment allows an increase in the reactor coolant system controlled leakage rate as shown in Specification 3.4.6.2.e, from 10 gpm to 12 gpm.

Date of issuance: January 23, 1986.

Effective date: January 23, 1986.

Amendment No.: 85.

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

Date of initial notice in Federal Register: November 20, 1985, 50 FR 47862. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 23, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room

Location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629.

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

Date of application for amendment: May 28, 1985, as supplemented September 30, 1985.

Brief description of amendment: This amendment revises Figure 6-2 in the TMI-1 Technical Specifications. Figure 6-2 is an organization chart titled "TMI-1 Unit Staff." The amendment adds the positions titled "Manager(s), Plant Engineering" which report to the Plant Engineering Director, and limits the number of managers in these positions to six. In addition, the amendment changes the "Chemistry Supervisor" block title to "Staff Chemist."

Date of issuance: January 15, 1986.

Effective date: January 15, 1986.

Amendment No.: 112.

Facility Operating License No. DPR-50: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 6, 1985, 50 FR 46214.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 15, 1986.

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 17126.

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

Date of application for amendment: July 31, 1985.

Brief description of amendment: The amendment updates Technical Specification Table 3.18-1, "Fire Detection Instrumentation," to include three locations where fire detection instrumentation has been added as a result of NRC acceptance of exemption requests.

Date of issuance: January 14, 1986.

Effective date: January 14, 1986.

Amendment No.: 111.

Facility Operating License No. DPR-50: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 28, 1985 (50 FR 34942).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 14, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, Pennsylvania 17126.

Indiana and Michigan Electric Company, Docket No. 50-316, Donald C. Cook Nuclear Plant, Unit No. 2, Berrien County, Michigan

Date of application for amendment: December 13, 1985, as supplemented December 19, 1985.

Brief description of amendment: The amendment extends on a one-time basis the 18 month surveillance frequency by 2 months for testing the reactor trip system instrumentation, the engineered safety feature actuation system

instrumentation, the containment sump level and flow monitoring instrumentation, the reactor coolant pump system relief and block valve instrumentation, the reactor coolant pump spray headers, the electrical power systems including: the alternate source, diesel generator and batteries, the energy core cooling system subsystem, some snubbers, and inspection of the divider barrier seal.

Date of issuance: January 28, 1986.

Effective date: January 28, 1986.

Amendment No.: 78.

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

Date of initial Notice in Federal Register: January 10, 1986 (51 FR 1315).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 28, 1986.

No significant hazards consideration comments received: No.

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

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

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

Date of application for Amendment: June 6, 1985.

Brief description of amendment: This amendment revises the Technical Specifications by adding four smoke detectors in the Reactor Auxiliary Building and by correcting the identifying numbers of 3 containment isolation valves.

Date of issuance: January 27, 1986.

Effective date: January 27, 1986.

Amendment No.: 3.

Facility Operating License No.: NPF-38.

Amendment revised the Technical Specifications.

Date of initial Notice in Federal Register: The Commission's related

evaluation is contained in a Safety Evaluation dated January 27, 1986.

No significant hazards consideration comments received: No.

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

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

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

Date of amendment request: October 17, 1985.

Brief description of amendment: Change Figure 6.2.1 to reflect a change in the management organization.

Date of issuance: January 16, 1986.

Effective date: January 16, 1986.

Amendment No.: 77

Facility Operating License No. DPR-63: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 4, 1985 (50 FR 49787).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 16, 1986.

No significant hazards consideration comments received: No.

Local Public Document Room location: State University College at Oswego, Penfield Library—Documents, Oswego, New York 13126.

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

Date of application for amendment: June 24, 1983.

Brief description of amendment: The amendment revises the Technical Specifications to provide limiting Conditions for Operation and Surveillance Requirements for the following items: (1) Overtime Limitations, (2) Reporting Safety/Relief Valve Failure and Challenges, (3) Reactor Core Isolation Cooling (RCIC) and RCIC Suction Transfer, (4) Isolation of RCIC Modifications and (5) Additional Monitoring Instrumentation. These changes relate to TMI Action Items covered by Generic Letters 83-02 and 83-36 dated January 10 and November 1, 1983 respectively. The items not included in this amendment either have been resolved or will be addressed separately.

Date of issuance: January 22, 1986.

Effective date: January 22, 1986.

Amendment No.: 37.

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

Date of initial notice in Federal Register: October 9, 1985 (50 FR 41250).

The Commission's related evaluation of the amendment is contained in a safety Evaluation dated January 22, 1986.

No significant hazards consideration comments received: No.

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

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.

Dates of application of amendments: November 27, 1985.

Brief description of amendments: The amendments change the Technical Specification 3/4.1.1.3, "Moderator Temperature Coefficient" to reflect the use of a more negative moderator temperature coefficient needed for end-of-cycle operations in Cycle 2.

Date of Issuance: 1/27/86.

Effective date: 1/27/86 and fully implemented within 30 days of issuance.

Amendment Nos.: 41 and 30.

Facility Operating License Nos. NPF-10 and NPF-15: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 27, 1985 (56 FR 53031).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated 1/27/86.

No significant hazards consideration comments were received.

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

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee.

Date of application of amendment: May 6, 1985.

Brief Description of amendment: The amendments change the Technical Specifications to require that acoustic monitors be one of the two required channels of pressurizer power relief valve and safety valve position indicators in accident monitoring tables and bases.

Date of Issuance: January 29, 1986.

Effective date: January 29, 1986

Amendment Nos.: 43 and 35.

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 31, 1985 (50 FR 31072).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 1986

No Significant Hazards Consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Bicentennial Library, 1001 Broad Street, Chattanooga, Tennessee 37401.

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

Date of application for amendment: August 13, 1985 as supplemented by letter dated November 15, 1985.

Brief description of amendment: Technical Specification 4.7.10.1.2.c has been revised to allow the 18-month inspection of a fire pump diesel engine to be performed when the plant is at power, as well as when the plant is shutdown. By letter dated November 15, 1985, the licensee informed us that the subject inspection will be performed either during shutdown or during power operation when the other two fire pumps are operable. By the November 15, 1985 letter, the licensee proposed wording for Specification 4.7.10.1.2.c to clarify this point.

Date of issuance: January 22, 1986.

Effective date: January 22, 1986.

Amendment No.: 11.

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

Date of initial notice in Federal Register: September 25, 1985 (50 FR 38923).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 22, 1986.

No significant hazards consideration comments received: No

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

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

Date of amendment request: October 15, 1985, as supplemented by letter dated December 23, 1985.

Description of amendment request: The amendment revises Technical Specification Figures 3.9-1 and 5.6-1 with curves that represent criteria for storing Westinghouse optimized fuel or standard fuel in Region 2 of the spent fuel pool, revises the maximum initial enrichment limit for reload fuel in the reactor and for storage of reload fuel in the spent fuel pool from 3.5 weight percent uranium-235 to 4.2 weight percent uranium-235, and revises the nominal center-to-center distance between fuel assemblies placed in storage racks from 9.14 to 9.24 inches.

Date of issuance: January 24, 1986

Effective date: January 24, 1986

Amendment No.: 12

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

Date of initial notice in Federal Register: November 6, 1985 (50 FR 46218) as corrected by notice on December 2, 1985 (50 FR 49468)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 24, 1986.

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.

Dated at Bethesda, Maryland, this 5th day of February 1986.

For the Nuclear Regulatory Commission.

Steven A. Varga,

Acting Deputy Director, Division of PWR Licensing-A Office of Nuclear Reactor Regulation.

[FR Doc. 86-2881 Filed 2-11-86; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-498A]

Houston Lighting and Power Co. et al. Receipt of Antitrust Information

The Houston Lighting and Power Company acting as agent for the City of Austin, the City Public Service Board of San Antonio, and Central Power and Light Company has submitted antitrust information in conjunction with the application for an operating license for a pressurized water reactor, known as South Texas, Unit 1, located in Matagorda County, Texas, 15 miles southwest of Bay City. The data submitted contain antitrust information for review, pursuant to NRC Regulatory Guide 9.3, necessary to determine whether there have been any significant changes since the antitrust settlement in September of 1980.

On completion of a staff antitrust review, the Director of Nuclear Reactor Regulation will issue an initial finding as to whether there have been "significant changes" under section 105c(2) of the Atomic Energy Act. A copy of this finding will be published in the Federal Register and will be sent to the Washington, DC and local public document rooms and to those persons providing comments or information in response to this notice. If the initial finding concludes that there have not