bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to William D. Beckner: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC (20005–3502), 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated May 9, 1996, as supplemented by letter dated August 27, 1996, which are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

Dated at Rockville, Maryland, this 4th day of September, 1996.

For the Nuclear Regulatory Commission.

Jack N. Donohew,

Senior Project Manager, Project Directorate IV–1, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–23192 Filed 9–10–96; 8:45 am]

BILLING CODE 7590–01–P

National State Liaison Officers' Meeting

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: The Nuclear Regulatory Commission (NRC) will sponsor a national meeting on October 8 and 9, 1996 with the State Liaison Officers to discuss items of mutual regulatory interest. The State Liaison Officers are appointed by the Governors of the fifty States and the Commonwealth of Puerto Rico to provide a communication channel between the States and the NRC.

DATES: The public meeting will be held on Tuesday, October 8, 1996 from 8:00 a.m. to 5:00 p.m.; Wednesday, October 9, 1996 from 8:30 a.m. to 12:00 noon.

ADDRESSES: The meeting is to be held at the NRC's Two White Flint Building Auditorium, 11554 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Spiros C. Droggitis, Office of State Programs, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 415–2367, FAX (301) 415–3502 & Internet (SCD@NRC.GOV).

SUPPLEMENTAL INFORMATION: Potential topics of discussion will include: status of NRC's Strategic Assessment and Rebaseling effort; current nuclear power plant issues; electric utility industry restructuring and economic deregulation; regulatory reform of radiation in medicine; external regulation of the U.S. Department of Energy; high-level radioactive waste, spent fuel storage and transportation issues; NRC's enforcement policy; and emergency planning and response issues.

The meeting will be conducted in a manner that will expedite the orderly conduct of business. The following procedures apply to public attendance at the meeting:

1. Questions or statements from attendees other than participants, i.e., other participating representatives of States and participating NRC staff will be entertained as time permits; and
2. Seating for the public will be on a first-come, first-served basis.

Dated at Rockville, Maryland this 5th day of September, 1996.

For the Nuclear Regulatory Commission.

Richard L. Bangart,

Director, Office of State Programs.

[FR Doc. 96–23191 Filed 9–10–96; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 19, 1996, through August 29, 1996. The last biweekly notice was published on August 28, 1996 (61 FR 44353).

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

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

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

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

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

By October 11, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendments request: August 1, 1996

Description of amendments request: The amendment will allow use of blind flanges during MODES 1-4 in the Calvert Cliffs Units 1 and 2 Containment Purge Systems. These flanges will establish integrity in Mode 5, prior to entering Mode 4, and maintain it in Modes 1-4, functions presently served by the valve.

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

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

The purpose of the Containment Purge System is to provide ventilation for the containment while in a shutdown condition. Valves, which are disabled in the shut position in Modes 1-4, may be opened in Modes 5 and 6 to allow air flow, are provided in the supply and exhaust piping, and are automatically shut on a Containment Radiation Signal to prevent release of radioactive material in the event of a fuel handling incident. Manual operation is also provided. In Modes 1-4, the valves are kept shut to provide containment integrity to withstand a presumed increase in containment pressure in the event of a loss-of-coolant accident. The proposed change will allow blind flanges to serve in place of the purge valves in Modes 1-4 by blocking off the purge penetration on both the supply and exhaust sides. The blind flanges will provide the same level of containment integrity previously provided by the purge valves. The revised Technical Specifications will continue to verify containment building leakage is maintained within the allowable limits by requiring the performance of a 10 CFR Part 50, Appendix J, Type B, leakage test on the blind flanges. The outside valve in each containment purge penetration will be removed and the inside valves will be left in place. The remaining inside valves will no longer by required to provide containment integrity in Modes 1-4. Only one of each pair of valves was credited for containment closure (Modes 5 and 6); therefore, removing the outside valves and the associated automatic closure signals is not a modification of the required capability to close the penetration. The inside valves will maintain their current safety function to close containment (if needed) by closing either on a Containment Radiation Signal (Mode 6) or manually (Modes 5 and 6). The Technical Specification surveillances associated with the purge valves will be changed to reflect the proposed modification to the plant. Since the blind flanges will limit radiological releases in Modes 1-4, and the purge valves will limit radiological releases in Modes 5 and 6, the proposed change will not increase the consequences of an accident previously evaluated.

The Containment Purge System is not an accident initiator but acts to limit the consequences of accidents. The system will provide containment isolation in Modes 1-4 as before, and the inside valves will still be available to close in Modes 5 and 6. Therefore, the proposed change does not increase the probability of an accident previously evaluated.

As stated above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

This requested change does not involve a significant alteration of the operation of the plant, and no new accident initiation mechanism is created by the modification. Four purge valves per unit currently provide containment closure in Modes 5 and 6. The outside valve in the supply and the exhaust lines will be removed to allow for installation of a blind flange in each line. The remaining supply and exhaust valves inside containment will continue to provide containment closure. The function currently performed by the four purge valves in Modes 1, 2, 3 and 4 will be performed by the blind flanges. Other, similar, blind flanges have been in service in the plant for a number of years, and have proven reliable. The Technical Specification surveillances associated with the testing of the purge valves and flanges will be changed to reflect the proposed modification to the plant. Therefore, this change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in the margin of safety.

The valves in the Containment Purge System currently provide containment integrity during Modes 1, 2, 3 and 4, and containment closure during Modes 5 and 6. The function currently performed by the purge valves in Modes 1, 2, 3 and 4 will be performed by the blind flanges. Because of their design and mounting method, the blind flanges will perform the containment integrity function as well as, or better than, the purge valves. In Modes 1-4, the double o-rings in the blind flanges will provide single-failure protection similar to the other existing Type B penetrations. The established allowable containment building leakage rate will be maintained by the implementation of a requirement to perform 10 CFR Part 50, Appendix J, Type B, leakage rate on the installed blind flanges. The outside valve in each purge containment penetration will be removed. Single failure is not assumed in the fuel handling accident analysis, therefore, removing the outside valves and their Containment Radiation Signal channels is not a modification of the required capability to close the penetration. The remaining inside valves will continue to provide automatic and manual containment closure in Mode 6 to mitigate the effects of a fuel handling accident. The Technical Specification surveillances associated with purge valve testing will be changed to reflect the proposed modification to the plant. Therefore, this change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

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

NRC Project Director: Jocelyn A. Mitchell, Acting Director

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: August 7, 1996

Brief description of amendment: The amendment proposes revising the Technical Specifications (TSs) to allow the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing. This performance-based Option B may be used as an alternative to the requirements in Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' of 10 CFR Part 50. To implement Option B to Appendix J, the amendment proposes modifying TSs to eliminate reference to the prescriptive Appendix J requirements and instead reference NRC Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' The amendment also proposes an editorial correction to the mathematical formula minimum testing frequency in the basis for TS 4.1.

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

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

For Indian Point Unit No. 2, the integrated leak rate testing [ILRT] as-found measured leakage rate acceptance criteria is changed from 0.75 La to 1.0 La. This change is consistent with the revised 10 CFR 50 Appendix J, NEI 94-01, ''Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.'' In addition, an as-found leakage rate acceptance criteria of 1.0 LA for Type A tests is consistent with the design basis and accident analysis assumptions. The as-left acceptance criteria remains unchanged at 0.75 La in accordance with the NEI guidance. Therefore, prior to entering an operating mode where containment integrity is required the as-left leakage rate will not exceed 0.75 La. The combined leakage rate for containment isolation valves listed in Technical Specification Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing, and portions of the sensitive leakage rate test which pertain to containment penetrations and double-gasketed seals shall be less than 0.6 La. The extensive operations and testing experience derived from industry show that risk to the general population is generally insensitive to changes in the allowable leakage rate. It has been determined that the allowable containment leakage can be increased by one to two orders of magnitude without significantly impacting the estimates of population dose in the event of an accident. Furthermore, the Indian Point Unit No. 2 ILRT test history provides substantial justification for the proposed changes.

Test results demonstrate that IP2 [Indian Point 2] has a low leakage containment and that the proposed changes would not jeopardize the ability of the containment to maintain the leakage rate at or below the required limits. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the probability and the consequence of a design basis accident are not being increased by the proposed changes.

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

Plant systems and components will not be operated in a different manner as a result of the proposed Technical Specification change. The proposed change permits a performance-based approach to determining the leakage-rate test frequency for the containment and containment penetrations (Type A, B, and C tests). There are no plant modifications, or changes in methods of operation. Therefore, the changes in testing intervals for the containment and containment penetrations have no affect on the probability of occurrence of a LOCA [loss-of-coolant-accident]. The Limiting Conditions for Operation are not being changed. Changing the as-found leakage-rate acceptance criterion to 1.0 La does not increase the probability or consequences of an accident. Changing the test interval for the containment and containment penetrations does not create any new accident precursors or methods of operation. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the possibility for an accident of a different type than was previously evaluated in the safety analysis report is not created by the proposed Technical Specification.

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

While the proposed changes do increase the probability for malfunction of equipment important to safety due to the longer intervals between leakage tests, it has been estimated that the longer test intervals will have an insignificant increase in the overall accident risk to the public. This increase has been reviewed and found to be acceptable by the NRC as documented in NUREG-1493 and the recent rulemaking to 10 CFR 50 Appendix J. We also agree that this increase in accident risk is insignificant. Changing the as-found acceptance criterion to 1.0 La does not increase the consequences of an accident, since the accident analysis assume a leakage rate of La for design basis accidents. The as-left Type A test acceptance criterion remains at less than 0.75 La. Given that the Indian Point Unit No. 2 ILRT test history show no failures during plant life, the proposed changes should not lead to a significant probability of creating new leakage paths or increased leakage rates. The proposed change to Technical Specification 4.1 Basis represent a minor editorial correction to the mathematical formula for minimum testing frequency which does not change the formula. Therefore, the accident analysis assumptions for design basis accidents are unaffected and the margin of safety is not decreased by the proposed Technical Specification change.

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

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

Date of amendment request: January 18, 1996

Description of amendment request: The proposed amendment would delete the requirement to perform inservice inspections of the primary coolant pump (PCP) flywheels.

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

The following evaluation supports the finding that operation of the facility in accordance with the proposed change to the Technical Specifications would not:

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

The proposed change to the Technical Specifications would delete the requirement to perform non-destructive examination of the upper flywheel on the PCPs. The fracture mechanics analyses conducted to support the change show that a preexisting crack sized just below detection level will not grow to the flaw size necessary to result in flywheel failure within the life of the plant. This analysis conservatively assumes minimum material properties, maximum flywheel accident speed, location of the flaw in the highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the allowable flaw size under normal operating conditions or to the critical flaw size under LOCA [loss-of-coolant accident] conditions over the life of the plant, elimination of inservice inspection for such cracks during the plant's life will not involve a significant increase in the probability of an accident previously considered.

The proposed changes do not increase the amount of radioactive material available for release or modify any systems used for mitigation of such releases during accident conditions. Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the Technical Specifications would not change the design, configuration, or method of operation of the plant and therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change to the Technical Specifications would not result in a significant reduction in the margin of safety. Significant conservatisms have been used for calculating the allowable flaw size, critical flaw size and crack growth rate in the PCP flywheels. These include minimum material properties, maximum flywheel accident speed, location of the postulated flaw in highest stress area and a number of startup/shutdown cycles eight times greater than expected. Since an existing flaw in the flywheel will not grow to the maximum allowable flaw size under normal operating conditions or to the critical flaw size under LOCA conditions over the life of the plant, elimination of inservice inspections for such cracks during the plant's life will not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John Hannon

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

Date of amendment request: August 8, 1996

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) of each unit to reference updated or recently approved methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report (COLR).

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

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

The proposed changes are administrative in nature, and do not affect any system, procedure, or manipulation of any equipment which could affect the probability or consequences of any accident.

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

The proposed changes are administrative in nature, and cannot introduce any new failure mode or transient which could create any accident.

(3) The proposed changes will not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature, and will not affect any operating parameters or limits which could result in a reduction in a margin of safety.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: June 21, 1996

Description of amendment request: The proposed amendments would revise the term ''lifting loads'' used in Technical Specification 3.9.6b.2, Manipulator Crane, to ''lifting force.'' This revision will clarify that the static loads associated with the lifting tool, drive rod and control rod weights are not included in the lifting force limit.

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

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

No. The proposed change is administrative in nature and does not represent any changes to the refueling process in the field. It more accurately describes the components for which the LCO's [Limiting Condition for Operation] protection is intended as well as giving a more accurate description of the auxiliary hoist's minimum capacity. It also broadens the domain of activities for which protective measures are taken by including drag load testing into monitored activities. At CNS [Catawba Nuclear Station], the auxiliary hoists and the manipulator cranes are rated at [greater than or equal to] 3000 pounds and are surveillance tested to greater than 1000 pounds. This brackets the limit force lifting value change from 600 to 1000 pounds in the amendment proposal.

Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This proposed administrative change reflects no changes in the refueling processes, or any systems, structures or components connected with the refueling process.

Will the change involve a significant reduction in a margin of safety?

No. The proposed administrative change has no impact on refueling processes, systems, structures or components, and does not result in any significant reduction in a margin of safety. The subject change only clarifies the original intent of the specification and more accurately describes the involved components, component capacities and the domain of activities for which measures are taken to protect the reactor internals.

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

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

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

NRC Project Director: Herbert N. Berkow

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

Date of amendment request: August 23, 1996 (TSCR 245)

Description of amendment request: The amendment request proposes new pressure-temperature (P-T) limits up to

22, 27, and 32 effective full power years (EFPY). The new sets of P-T curves would be used beyond 17 EFPY in the future as the corresponding EFPY of operation is completed.

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

We have determined that this change request with respect to P-T limits involves no significant hazards considerations in that operation of the Oyster Creek Plant in accordance with the proposed amendment, will not:

1. Involve a significant increase in the probability of an accident because the new limits account for the increase in RT $_{NDT}$, including statistical uncertainty, due to neutron irradiation of the reactor vessel as well as establishing initial RT $_{NDT}$ on the basis of current Code requirements, also including statistical uncertainty, in accordance with Reg. Guide 1.99, Rev. 2. The new P-T curves will assure that brittle fracture of the reactor vessel is prevented.

2. Create the probability of a new or different kind of accident from any accident previously evaluated. These new limits are the result of the calculation methodology in Reg. Guide 1.99, Rev. 2 [Radiation Embrittlement of Reactor Vessel Materials], as required by Generic Letter 88-11 [NRC Position on Radiation Embrittlement of Reactor Materials and its Impact on Plant Operations]. Primary system configuration and function remain unchanged.

3. Involve a significant reduction in margin of safety because the bases for the margin of safety remain the same as current limits, i.e., ASME [American Society of Mechanical Engineers], Sect. XI, App. G for available fracture toughness and applied stress intensity, Reg. Guide 1.99, Rev. 2 for calculating applied stress intensity, Reg. Guide 1.99, Rev. 2 for calculating adjusted RT $_{NDT}$ and 10 CFR 50, App. G, for criticality conditions.

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

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

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

NRC Project Director: John F. Stolz

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: August 15, 1996

Description of amendment request: The proposed amendment would modify the Clinton Power Station Technical Specifications to incorporate the revised Safety Limit Minimum Critical Power Ratio (SLMCPR) as calculated by General Electric (GE) for Cycle 7 operation. The need to change the SLMCPR resulted from the 10 CFR Part 21 condition reported by GE in their letter to the NRC dated May 24, 1996.

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

(1) This change does not involve a significant increase in the probability or consequences of any accident previously evaluated. In lieu of utilizing a potentially nonconservative generic value, this change revises the SLMCPR to be appropriately conservative as it has been specifically calculated on a plant- and cycle-specific basis. Although the SLMCPR does not apply (i.e., is not assumed or required to be met) during any analyzed accident, the MCPR fuel cladding Safety Limit ensures that during normal operation and during anticipated operational occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling. The revised value for the SLMCPR is determined using the same methodology as the previous SLMCPR with the exception that it utilizes plant specific conditions to determine the safety limit. The revised SLMCPR, therefore, accounts for actual expected power distributions in the Clinton Power Station (CPS) core as well as CPS-specific uncertainties. This provides a more conservative SLMCPR than the generic value used previously.

The proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents previously evaluated. In addition, the proposed change does not affect the ability of any plant systems or equipment to operate as assumed in the safety analyses. The revised SLMCPR will continue to ensure that the fuel cladding integrity is not lost as a result of over-heating during normal plant operation or any AOO. As a result, the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

(2) The proposed change does not involve any new modes or operation, any changes to setpoints, or any plant modifications. Further, the incorporation of a revised MCPR safety limit, which has been determined to be acceptable for CPS Cycle 7 operation, does not result in the creation of any new failure modes or potential precursors to an accident. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

(3) The proposed SLMCPR has been evaluated to ensure that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling. As noted above, the revised SLMCPR has been determined using the same methodology as used previously with the exception of using CPS Cycle 7 specific core and fuel design data. This change ensures that the margin of safety for fuel cladding integrity is maintained by providing a CPS specific MCPR safety limit as opposed to utilizing a potentially less conservative generic limit. Therefore, the implementation of the proposed change to the SLMCPR does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

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

Date of amendment requests: May 8, 1996

Description of amendment requests: The licensee proposes to revise improved Technical Specifications (TS) 3.9.4 and 3.9.5 to facilitate testing of low pressure safety injection system components and permit additional flexibility in scheduling maintenance on the shutdown cooling system.

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

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

Limiting Conditions for Operation (LCO) in Technical Specifications (TSs) 3.9.4 and 3.9.5 define the operability requirements for the Shutdown Cooling (SDC) system during refueling operations (Mode 6) while the water level above the top of the reactor vessel flange is at least 23 feet and less than 23 feet, respectively. The objective of these TSs is to ensure that 1) sufficient cooling is available to remove decay heat, 2) the water in the reactor vessel is maintained below 140°F, and 3) sufficient coolant circulation is maintained in the reactor core to minimize boron stratification leading to a boron dilution incident.

The proposed TS changes affect the current limits imposed while ensuring adherence to the bases of the TS. No plant modifications are being made. The reactor cavity water level limitations and SDC system required operating times are being changed based on plant specific calculations and the objectives of the TSs are being maintained.

1) Reduce the water level where two loops of SDC are required from 23 feet to 20 feet above the reactor vessel flange,

Prior to the approval of Unit 2 Amendment No. 127 and Unit 3 Amendment No. 116, Technical Specification Bases Section 3/4.9.8 has stated that ''With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.''

In the Bases for the New Standard Technical Specifications, ''NUREG 1432, Revision 0, dated September 30, 1992, Section B 3.9.4 it is stated that; ''The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, ''Refueling Water Level.''

Southern California Edison (Edison) calculations show that there is an insignificant difference in the time to boil due to the 3-foot change in required water level. Therefore, adequate water is still available to mitigate the consequences of losing SDC.

2) Increase the time a required loop of the SDC system may be removed from service from up to 1 hour per 8-hour period to up to 2 hours per 8-hour period, provided the upper guide structure has been removed from the reactor vessel,

The proposed TS changes the time the SDC loop may be removed from operation from up to 1 hour per 8-hour period to up to 2 hours per 8-hour period, and allows removal of the SDC loop from operation for testing of the Low Pressure Safety Injection (LPSI) system components as well as for core alterations in the vicinity of the hot legs. The proposed TS change also imposes certain restrictions to ensure operating the SDC system in accordance with this proposed TS change is of no safety significance. These [r]estrictions are discussed separately below.

Specifically stating that the upper guide structure will be removed assures that natural heat transfer is not impeded.

When securing the only operating loop of the SDC system the maximum Reactor Coolant System (RCS) temperature is maintained [less than or equal to] 140°F. The initial conditions and heatup rate are selected such that the RCS temperature remains [less than or equal to] 140°F during the test. Therefore, there is ample margin to boiling. Typical initial temperatures are less than 100°F.

The water being injected by the LPSI system test is cool water from the Refueling Water Storage Tank (RWST) and will increase the reactor cavity water level by several inches, providing more cool water to the heat sink. The two hours is sufficient time to align the system to test, perform the test, and restore the loop of SDC to operation prior to exceeding 140°F.

No operations are permitted that would cause a reduction of the RCS boron concentration. This minimizes the probability of an inadvertent boron dilution event. The use of adequately borated water for injection into the RCS during the test provides assurance that the test itself cannot lead to a boron dilution event. When the SDC system is operating, the minimum SDC flow rate of 2200 gpm imposed by Surveillance Requirements SR 3.9.4.1 and SR 3.9.5.1 is sufficient to ensure complete mixing of the boron within the RCS.

Securing SDC flow is only allowed when the reactor cavity water level is maintained greater than or equal to 20 feet above the reactor vessel flange. This level ensures an adequate heat sink to perform the LPSI pump suction header check valve test.

3) Allow for running 1 loop of shutdown cooling with additional requirements when the water level is less than 20 feet but greater than or equal to 12 feet above the reactor vessel flange,

4) Add an action to be taken when operating 1 loop of SDC with less than 20 feet of water above the reactor vessel flange when the specified requirements are not met,

In the event of a loss of SDC the time to boil is reduced from approximately 4.0 hours when the water level is 23 feet above the reactor vessel flange to approximately 2.3 hours at 12 feet, assuming the reactor has only been shutdown for 6 days. However, this is ample time to close containment (less than 1 hour) and to restore SDC or initiate alternative cooling (e.g., add water to the cavity (approximately 1 hour)). The reactor pressure vessel flange is approximately 11' above the top of the fuel. Therefore, the water level will be a minimum of 23' above the fuel, which still maintains a large volume of water to provide a heat sink.

Requiring the reactor to be shutdown for at least 6 days to have only one loop of SDC operable when the reactor cavity level is between 20 feet and 12 feet above the reactor vessel flange ensures that the time to boil is greater than twice the time it would take to establish containment closure and to commence reactor cavity fill with the required standby equipment.

One loop of SDC operating with a containment spray pump allows for the high capacity LPSI pump to be the main standby pump capable of filling the reactor cavity to at least 20 feet above the reactor pressure vessel flange in the event SDC is lost. The high pressure safety injection pump will also be maintained OPERABLE to increase the water level if needed. In support of this contingency the RWST will be required to contain the volume of water needed to raised [raise] the level to 20 feet above the reactor pressure vessel flange. As discussed above, the reactor cavity can be filled at a rate of approximately 4.0 inches per minute with the LPSI pump.

If operating one loop of the SDC system with less than 20 feet of water above the reactor vessel flange and any of the required conditions are not met, requiring immediate action to establish greater than or equal to 20 feet of water above the reactor vessel flange ensures no time is wasted trying to restore the required condition not met. By taking action to restore the level to 20 feet above the reactor vessel flange the plant will be placed in TS 3.9.4, which only requires one loop of SDC to be operable. Additionally, the core will not heat up while the water level in the reactor cavity is being raised with cool water from the RWST. This will provide additional time to either restore the one loop of SDC or take other actions to provide core cooling as required by TS 3.9.4.

A Probabilistic Risk Assessment (PRA), with a) one loop of the SDC system operable with the reactor cavity water level greater than or equal to 12 feet above the reactor vessel flange, and b) one loop of the SDC system operable with the reactor cavity water level greater than or equal to 20 feet above the reactor vessel flange, showed that the operations in accordance with the proposed TS would not significantly increase the probabilities of inventory boiling and core damage.

5) Item 6 adds wording to the notes in LCOs 3.9.4 and 3.9.5 that was unintentionally deleted by the Unit 2 Amendment No. 127 and Unit 3 Amendment No. 116.

This is an editorial change.

Therefore, proposed changes 1 through 5 do not involve a significant increase in the probability or consequences of an accident.

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

1) Reduce the water level where two loops of SDC are required from 23 feet to 20 feet above the reactor vessel flange,

2) Increase the time a required loop of the SDC system may be removed from service from up to 1 hour per 8-hour period to up to 2 hours per 8-hour period, provided the upper guide structure has been removed from the reactor vessel,

3) Allow for running 1 loop of shutdown cooling with additional requirements when the water level is less than 20 feet but greater than or equal to 12 feet above the reactor vessel flange,

4) Add an action to be taken when operating 1 loop of SDC with less than 20 feet of water above the reactor vessel flange when the specified requirements are not met,

The Limiting Conditions for Operation (LCO) in Technical Specifications (TSs) 3.9.4 and 3.9.5 define the operability requirements for the SDC system during refueling operations (Mode 6) while the water level above the top of the reactor vessel flange is at least 23 feet and less than 23 feet, respectively. The objective of the proposed TS changes is to ensure that the intent of the Bases is maintained. [i.e., 1) sufficient cooling is available to remove decay heat, 2) water in the reactor vessel is maintained below 140°F, and 3) sufficient coolant circulation is maintained in the reactor core to minimize boron stratification leading to a boron dilution incident.]

The proposed TS changes affect the current limits imposed while ensuring adherence to the bases of the TS. No plant modifications are being made. The reactor cavity water level limitations and SDC system required operating times are being changed based on plant specific calculations, and the objective of the TSs are being maintained. The added requirements and action statement facilitate safe operation.

5) Item 6 adds wording to the notes in LCOs 3.9.4 and 3.9.5 that was unintentionally deleted by the Unit 2 Amendment No. 127 and Unit 3 Amendment No. 116.

This is an editorial change.

Therefore, the operation of the facility in accordance with proposed changes 1 through 5 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Limiting Conditions for Operation (LCO) in TSs 3.9.4 and 3.9.5 define the operability requirements for the SDC system during refueling operations (Mode 6) while the water level above the top of the reactor vessel flange is at least 23 feet and less than 23 feet, respectively. The objectives of these TSs are to ensure that 1) sufficient cooling is available to remove decay heat, 2) the water in the reactor vessel is maintained below 140°F, and 3) sufficient coolant circulation is maintained in the reactor core to minimize boron stratification leading to a boron dilution incident.

1) Reduce the water level where two loops of SDC are required from 23 feet to 20 feet above the reactor vessel flange,

Prior to the approval of Unit 2 Amendment No. 127 and Unit 3 Amendment No. 116, Technical Specification Bases Section 3/4.9.8 has stated that ''With the reactor vessel head removed and 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.''

In the Bases for the New Standard Technical Specifications, NUREG 1432, Revision 0, dated September 30, 1992, Section B 3.9.4 it is stated that ''The 23 ft level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, ''Refueling Water Level.''

Edison calculations show that there is a minimal difference in the time to boil due to the 3-foot change in required water level. Therefore, the margin of safety has not been significantly reduced.

2) Increase the time a required loop of the SDC system may be removed from service from up to 1 hour per 8-hour period to up to 2 hours per 8-hour period, provided the upper guide structure has been removed from the reactor vessel,

The proposed TS changes the time the SDC loop may be removed from operation from up to 1 hour per 8-hour period to up to 2 hours per 8-hour period, and allows removal of the SDC loop from operation for testing of the LPSI system components as well as for core alterations in the vicinity of the hot legs. The proposed TS change also imposes certain restrictions to ensure operating the SDC system in accordance with this proposed TS change is of no safety significance. These restrictions are discussed separately below.

Specifically stating that the upper guide structure will be removed assures that natural heat transfer is not impeded.

When securing the only operating loop of the SDC system, the maximum RCS temperature is maintained [less than or equal to] 140°F. The initial conditions and heatup rate are selected such that RCS temperature remains [less than or equal to] 140°F during the test. Therefore, there is ample margin to boiling. Typical initial temperatures are less than 100°F.

The water being injected by the LPSI system test is cool borated water from the RWST and will increase the level of the reactor cavity by several inches. The two hours is sufficient time to align the system to test, perform the test, and restore the loop of SDC to operation prior to exceeding 140°F.

No operations are permitted that would cause a reduction of the RCS boron concentration. This minimizes the probability of an inadvertent boron dilution event. The use of adequately borated water for injection into the RCS during the test provides assurance that the test itself cannot lead to a boron dilution event. When the SDC system is operating, the minimum SDC flow rate of 2200 gpm is sufficient to ensure complete mixing of the boron within the RCS.

Securing SDC flow is only allowed when the reactor cavity water level is maintained greater than or equal to 20 feet above the reactor vessel flange. This level ensures an adequate heat sink to perform the LPSI pump suction header check valve test.

The added requirements and the nature of the test provide assurances that the water temperature will be maintained less than 140°F and that boron stratification is prevented.

3) Allow for running 1 loop of shutdown cooling with additional requirements when the water level is less than 20 feet but greater than or equal to 12 feet above the reactor vessel flange,

4) Add an action to be taken when operating 1 loop of SDC with less than 20 feet of water above the reactor vessel flange when the specified requirements are not met,

In the event of a loss of SDC, the time to boil is reduced from approximately 4.0 hours when the water level is 23 feet above the reactor vessel flange to approximately 2.3 hours at 12 feet, when the reactor has only been shutdown for 6 days. However, this is ample time to close containment (less than 1 hour), and to restore SDC or initiate alternative cooling (e.g., add water to the cavity (approximately 1 hour)).

Requiring the reactor to be shutdown for at least 6 days to have only one loop of SDC operable when the reactor cavity level is between 20 feet and 12 feet above the reactor vessel flange ensures that the time to boil is greater than twice the time it would take us to establish containment closure and to commence reactor cavity fill with the required standby equipment.

One loop of SDC operating with a containment spray pump allows for the high capacity LPSI pump to be the main standby pump capable of filling the reactor cavity to at least 20 feet above the reactor pressure vessel flange in the event SDC is lost. The high pressure safety injection pump will also be maintained OPERABLE to increase the water level if needed. In support of this contingency the RWST will be required to contain the volume of water needed to raised [raise] the level to 20 feet above the reactor pressure vessel flange. As discussed above, the reactor cavity can be filled at a rate of approximately 4.0 inches per minute with the LPSI pump.

If operating one loop of the SDC system with less than 20 feet of water above the reactor vessel flange and any of the required conditions are not met, requiring immediate action to establish greater than or equal to 20 feet of water above the reactor vessel flange ensures no time is wasted trying to restore the required condition not met. By taking action to restore the level to 20 feet above the reactor vessel flange the plant will be placed in TS 3.9.4, which only requires one loop of SDC to be operable. Additionally, the core will not heat up while the reactor cavity water level is being raised with cool water from the RWST. This will provide additional time to either restore the one loop of SDC or take other actions to provide core cooling as required by TS 3.9.4.

A PRA showed that operations in accordance with the proposed TS did not significantly increase the probabilities of inventory boiling and core damage.

5) Item 6 adds wording to the notes in LCOs 3.9.4 and 3.9.5 that was unintentionally deleted by the Unit 2 Amendment No. 127 and Unit 3 Amendment No. 116.

This is an editorial change.

Therefore, operation of the facility in accordance with proposed changes 1 through 5 do not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Science Library, University of California, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

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

Date of amendment requests: May 9, 1996, as supplemented by letter dated June 27, 1996.

Description of amendment requests: The licensee proposes to add a requirement to maintain a Barrier Control Program to Section 5 of the improved Technical Specifications.

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

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

The proposed change will allow a passive support system, plant barriers, to be taken out of service for a specific allowed outage time. Since the allowed outage times are to limit the average annual cumulative increase in fuel damage risk to less than 1.0E-6, there will not be a significant increase in either the probability or consequences of any accident previously evaluated. Additionally, the proposed change will allow barrier impairments if allowed by a 10 CFR 50.59 evaluation and also if the equipment is declared inoperable or is not needed. Since these two conditions are already a part of the San Onofre Units 2 and 3 Licensing Basis, there will be no change in the probability or consequences of an accident previously evaluated.

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

Barriers have been analyzed for specific hazards. The nature of these hazards will not change due to this amendment, and therefore no new or different kind of accident will be created from any accident previously evaluated.

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

Since allowing barrier impairments in accordance with 10 CFR 50.59 or declaring affected equipment inoperable is part of the SONGS Units 2 and 3 Licensing Basis, there will be no reduction in the margin of safety from these two criteria.

Allowing allowed outage times for barrier impairments does not have a significant effect on a margin of safety because the average annual cumulative increase in fuel damage risk is limited to less than 1.0E-6/yr. This small increase is about 3% of the San Onofre Units 2 and 3 core damage risk as reported in the Individual Plant Examination (IPE).

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

Local Public Document Room location: Science Library, University of California, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

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

Date of amendment requests: May 29, 1996

Description of amendment requests: The licensee proposes to revise the acceptance criteria for the Agastat time delay relays used in the engineered safety features (ESF) load sequencer in Surveillance Requirement (SR) 3.8.1.18, ''A.C. Sources - Operating'' of Technical Specification (TS) 3.8.1, ''A.C. Sources - Operating.''

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

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

The proposed change would expand the current surveillance acceptance criteria to more accurately reflect the characteristics of the installed plant equipment. The diesel generators (DG's) have sufficient capacity to maintain adequate voltage and frequency during load sequencing with the expanded tolerance. The overall Engineered Safety Features (ESF) response times in the Technical Specifications and safety analyses are maintained even though the timer tolerance is increased, therefore, the consequences of any accident previously evaluated are not increased. The DG load sequence timers are not of themselves a credible initiator of any accident, so the probability of an accident has not been increased. The timers will function acceptably to support the equipment needed for accident mitigation, so the consequences of an accident are not increased. Therefore, the probability or consequences of any accident previously evaluated is not increased.

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

This amendment request does not involve any change to plant equipment or operation. In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DG's in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA). Increasing the timer tolerance will not create the possibility of a new or different kind of accident from any previously evaluated.

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

This amendment does not change the manner in which safety limits, limiting safety settings, or limiting conditions for operations are determined. The actual response times have not been altered by this amendment, therefore, operations will not be affected. Accordingly, this amendment will not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Science Library, University of California, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

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

Date of amendment requests: May 30, 1996

Description of amendment requests: The licensee proposes to revise Surveillance Requirements (SR) 3.6.1.1, 3.6.2.1, and 3.6.3.6, of the improved Technical Specifications. The proposed change will allow implementation of the recently approved Option B to 10 CFR Part 50, Appendix J. This new rule allows for a performance-based option for determining the test frequency for containment leakage rate testing.

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

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

Since the interval between containment leakage rate tests is not related in any way to conditions which cause accidents, and plant structures, systems, and components will not be operated in a different manner as a result of the proposed Technical Specification (TS) change, the proposed changes will not increase the probability of an accident previously evaluated.

Containment leakage may result from accidents which are evaluated in the Updated Final Safety Analysis Report. The proposed TS changes may result in an acceptably small increase in post-accident containment leakage. Using a statistical approach, NUREG-1493 determined that the increase in hypothetical dose to the public resulting from extending the testing interval is extremely small. NUREG-1493 concluded that such small hypothetical dose increases to the public are justifiable due to the real reduction in occupational exposure resulting from interval extension. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change only incorporates the performance based approach for containment leak rate testing authorized in the new Option B to Appendix J of 10 CFR Part 50. The interval extensions allowed, through this approach, do not have the potential for creating the possibility of new or different kinds of accidents from those previously evaluated because plant structures, systems, and components will not be operated in a different manner as a result of the TS change and, therefore, will not introduce any new or different failure modes or initiators. Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification does not alter the allowable containment leakage rate. The proposed change replaces the current, prescriptive testing requirements with a new performance based approach for establishing the testing intervals. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Local Public Document Room location: Science Library, University of California, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

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

Date of amendments request: August 23, 1996

Description of amendments request: The proposed amendments would revise the Technical Specifications to allow installation of laser welded elevated tubesheet sleeves in Farley, Units 1 and 2.

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

1. Operation of the Farley Nuclear Plant Units 1 and 2 steam generators in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The installation of elevated tubesheet laser welded sleeves as described below, can be used to repair degraded tubes by returning the condition of the tubes to their original design condition (for tube integrity, stress and fatigue considerations, and leaktightness during all plant conditions). Tube bundle overall structural and leakage integrity will be increased with the installation of the laser welded sleeves. The performance history of Westinghouse sleeves has shown that, to date, no domestic laser welded sleeves have been removed from service due to corrosion degradation of the sleeve or parent tube in the joint area.

Any hypothetical sleeve failure is bounded by the consequences of a postulated steam generator tube rupture event. The use of elevated tubesheet laser welded sleeves will not increase the amount of primary-to-secondary leakage anticipated during a postulated steam linebreak and other analyzed accidents. Leak rate tests show only negligible primary-to-secondary leakage through the non-welded elevated tubesheet sleeve lower joints during normal or accident conditions such that any consequences are insignificant with regard to offsite doses. Sleeve installation will result in an increase in resistance to primary coolant flow through the tube. Depending on the assumed steam generator tube rupture location, the primary coolant flow through the ruptured tube is reduced by the influence of sleeves installed below the break location, thereby reducing the consequences to the public due to a steam generator tube rupture event. Steam generator tube sleeving has as a basis that the analyzed steam generator tube plugging level and associated minimum measured flow rate, is not exceeded. Therefore, primary coolant flow area assumptions in the accident analyses are not affected and any consequences of a postulated loss of coolant accident would not be increased.

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

Installation of elevated tubesheet laser welded sleeves will increase the leaktightness of the tube bundle in addition to enhancing overall steam generator tube bundle integrity by isolating localized tube wall degradation. Isolation of the tube degradation is provided by attachment between the tube and sleeve at each end of the sleeve. Following the installation of the sleeves, steam generator tube integrity is restored to its original design bases.

Testing has shown that once installed, there is no mechanism for the sleeves to affect any portion of the steam generator other than the tubes in which they are installed. No other system or component connecting with the steam generator is adversely affected by the operation of the steam generator following installation of laser welded tube sleeves.

Structural analyses of the tube, sleeve and sleeve joints show the stress limits defined in the ASME [American Society of Mechanical Engineers] Code are not exceeded during all plant conditions. The effect of any hypothetical failure of the sleeve would be bounded by existing tube rupture analyses. No increase in leakage is anticipated during a postulated steam line break event. Therefore, operation of the steam generators following installation of elevated tubesheet laser welded sleeves in the tubes of the Farley steam generators will not result in an accident previously not analyzed in the FSAR [Final Safety Analysis Report].

Therefore, SNC [Southern Nuclear Operating Company] concludes that the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety with respect to maintenance of the integrity of the tube bundle is provided, in part, by the safety factors included in the ASME Code, and is not reduced. Nondestructive examination of the sleeve and non-sleeved tube length still can be performed; therefore, the recommendations of Regulatory Guide 1.83, Revision 1 can be implemented. The installation process of the elevated tubesheet laser welded sleeves has been shown to provide an essentially leaktight bond between the sleeve and the tube during all plant conditions, and, as such, would not significantly contribute to the radiological consequences of a postulated steam line break event. Any combination of sleeving and plugging utilized at Farley Units 1 and 2 up to the level that analyzed minimum measured reactor coolant flow rate is maintained per Technical Specification requirements, will be bounded by the accident analyses supporting the analyzed flow level.

Therefore, SNC, concludes that the proposed change does not result in a significant reduction in a loss of margin with respect to plant safety as defined in the Final Safety Analysis Report or the bases of the Farley technical specifications.

Based on the preceding analysis, it is concluded that operation of the Farley

Nuclear Plant steam generators in accordance with the proposed amendment does not involve a significant hazards consideration as defined in 10 CFR 50.92.

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

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

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

NRC Project Director: Herbert N. Berkow

Tennessee Valley Authority, Docket Nos. 50-390 Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: June 29, 1996 (TS 5.2.2.f)

Description of amendment request: The proposed amendment would revise the Watts Bar (WBN) Unit 1 Technical Specification (TS) requirements to delete the first sentence of TS Section 5.2.2.f which reads, ''The Operations Manager shall hold or have held an SRO [Senior Reactor Operator] license on a similar unit.'' The remaining sentence of this section is being revised to indicate that the Operations Superintendent will hold an SRO license for WBN Unit 1. This change is consistent with the Tennessee Valley Authority's (TVA) commitment to ANSI N18.1-1971 regarding the qualification of this position and is consistent with the Standard TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

Operation of the plant in accordance with the proposed amendment will not:

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

As explained in the June 29, 1996 submittal, the proposed change is considered to be administrative in nature. The proposed change affects an administrative control, which was based on the guidance of ANSI N18.-1971. ANSI N18.1-1971 recommended that the Operations Manager hold an SRO license. The ANSI N18.1-1971 Standard defines the positions of Plant Manager, Operations Manager, Supervisors and Operators. A subsequent update of this standard, ANSI/ANS 3.1-1987, also defines the position of Operations Middle Manager. The correlating named positions in the TVA management structure at WBN are: WBN Operations Manager correlates to ANSI Plant Manager, WBN Operations Superintendent correlates to ANSI Operations Manager or Operations Middle Manager, WBN Shift Operations Supervisor correlates to ANSI Shift Supervisor, and WBN Senior and Licensed Operators correlate to ANSI operators. The guidance in Section 4.2.2 of ANSI/ANS 3.1-1987 recommends that ''If the Operations Manager does not hold an NRC License, then the Operations Middle Manager shall hold an NRC Senior Operator's License. This would be consistent with TVA's proposal that the WBN Operations Superintendent (ANSI Operations Middle Manager) continue to be required to maintain an SRO license.

The proposed change does not alter the design of any system, structure, or component, nor does it change the way plant systems are operated. It does not reduce the knowledge, qualifications, or skills of licensed operators. The control room operators will continue to be supervised by the licensed Shift Supervisors and the first level of off-shift WBN managemet directing the activities of licensed operators will continue to hold an SRO license. In summary, the proposed change does not affect the ability of the Operations Superintendent to provide the plant oversight required of his position. Thus, it does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed change to TS 5.2.2.f does not affect the design or function of any plant system, structure, or component, nor does it change the way plant systems are operated. It does not affect the performance of NRC licensed operators. Operation of the plant will continue to be supervised by personnnel who hold an NRC SRO license. Based on the above, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change involves an administrative control. The proposed change does not reduce the level of knowledge or experience required of an individual who fills the Operations Superintendent position. The control room operators will continue to be supervised by personnel who hold an SRO license. Thus, the proposed change does not ivnolve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET llH, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

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

Date of amendment request: November 17, 1995, as supplemented July 29, 1996

Description of amendment request: The proposed amendment would revise Technical Specification Section 15.6.3, ''Facility Staff Qualifications.'' The title of the responsible health physicist would be changed, and a requirement for this individual to be a supervisor would be added.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of an accident previous evaluated.

The proposed changes separate the qualifications requirements of the Technical Specifications from the Health Physics Manager, while requiring that the same qualifications be fulfilled by a designated Health Physicist position within the organization. This change maintains the present knowledge requirements of the PBNP [Point Beach Nuclear Plant] staff. The personnel holding the health physics qualifications are not considered in the probability of any accident. By ensuring the appropriate expertise remains on the staff to advise management on issues related to radiological safety, appropriate action is assured during analyzed events to assess and mitigate the radiological consequences. Therefore, this change does not affect the probability or consequences of any accident previously evaluated.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a new or different kind of accident from any accident previously evaluated.

The proposed change separates the Health Physics Manager qualifications from the position while maintaining the requirements for that expertise to be maintained within the organization. This is an administrative change only and does not affect any plant structures, systems or components. Therefore, a new or different kind of accident from any accident previously evaluated cannot result.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant reduction in a margin of safety.

The proposed changes are administrative only. The required levels of expertise and experience will be maintained within the Health Physics organization. Therefore, there is no reduction in a margin of safety.

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

Local Public Document Room location: Joseph P. Mann Library, 1516 Sixteenth Street, Two Rivers, Wisconsin 54241

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

NRC Project Director: Gail H. Marcus

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

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

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

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

Date of amendment request: January 5, 1996, as supplemented July 12, 1996

Description of amendment request: The proposed amendment would revise the requirements of technical specification 3.1.9.3 to permit a filled refueling cavity to serve as a back-up means of decay heat removal.

Date of individual notice in the Federal Register: August 28, 1996 (61 FR 44348)

Expiration date of individual notice: September 27, 1996

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

Notice Of Issuance Of Amendments To Facility Operating Licenses

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

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

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

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

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

Date of application for amendments: March 15, 1995, as supplemented June 29, 1995, May 1, 1996 and May 15, 1996.

Brief description of amendments: The amendments revise the Technical Specification (TS) Section 6.0, ‘‘Administrative Controls’’ to be consistent with the guidance provided in the Improved Standard Technical Specifications (STSs) for Combustion Engineering Plants. Additionally, the amendments (a) allow the Shift Technical Advisory to perform dual roles, (b) establishes a TS Bases Control Program, (c) provides for a reduction in the reporting requirements, and (d) provides an option for estimating occupational doses.

Date of issuance: August 26, 1996

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

Amendment Nos.: 216 and 193

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

Date of initial notice in Federal Register: August 16, 1995 (60 FR 42598) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 26, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Connecticut Yankee Atomic Power Company, Docket No. 50-213, Haddam Neck Plant, Middlesex County and Northeast Nuclear Energy Company, et al., Docket Nos. 50-245, 50-336, and 50-423, Millstone Nuclear Power Station, Units 1, 2, and 3, New London County, Connecticut

Date of application for amendments: November 22, 1995

Brief description of amendments: The amendments replace the title-specific designation of members representing specific functional areas on the Plant Operating Review Committee (PORC) for the Haddam Neck Plant and Millstone Units 1, 2, and 3 with a functional area-specific designation that stipulates membership qualification and experience requirements. The amendments also clarify the composition of the Site Operations Review Committee (SORC) at Millstone.

Date of issuance: July 16, 1996

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

Amendment Nos.: 190, 95, 200, 130

Facility Operating License Nos. DPR-61, DPR-21, DPR-65, AND NPF-49: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7549) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 16, 1996 No significant hazards consideration comments received: No.

Local Public Document Room location: Russell Library, 123 Broad Street Middletown, Connecticut 06457, for the Haddam Neck Plant, and the Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut 06360, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut 06385, for Millstone 1, 2, and 3.

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

Date of application of amendments: June 6, 1996; supplemented August 1, 1996

Brief description of amendments: The amendments revise the Technical Specification requirements related to testing of the Low Pressure Service Water pumps and valves, LPSW-4 and LPSW-5, to reflect a design change to remove the Engineered Safeguards signal from the valves.

Date of Issuance: August 19, 1996

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

Amendment Nos.: 217, 217, 214

Facility Operating License Nos. DPR-38, DPR-47, and DPR-55: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37298) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 19, 1996. No significant hazards consideration comments received: No

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: May 31, 1996, as supplemented by letter dated May 2, 1996

Brief description of amendment: The amendment revised the schedule for withdrawing capsules with reactor vessel material specimens in accordance with the reactor vessel material surveillance program for the Grand Gulf Nuclear Station, Unit 1 and Section III.B.3 of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' of 10 CFR Part 50.

Date of issuance: August 21, 1996

Effective date: August 21, 1996

Amendment No: 127

Facility Operating License No. NPF-29: Amendment revises the license.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31179) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: May 17, 1995, as supplemented July 15, 1996.

Brief description of amendments: These amendments improve consistency between the Technical Specifications (TS) and the improved Combustion Engineering Standard Technical Specifications (STS) and resolve other inconsistencies in the TS.

Date of Issuance: August 14, 1996

Effective Date: August 14, 1996

Amendment Nos.: 146 and 85

Facility Operating License Nos. DPR-67 and NPF-16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 21, 1995 (60 FR 32363). The July 15, 1996, letter made a minor change to the proposed definition of core alteration which made it more closely match the wording in the STS and did not change the scope of the May 17, 1995, application and initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 14, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

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

Date of application for amendments: August 16, 1996

Brief description of amendments: Relocates selected Technical Specifications (TS) related to instrumentation to the Updated Final Safety Analysis Report, in accordance with the Commissions Final Policy Statement on TS Improvement for Nuclear Power Reactors (58 FR 39132, July 22, 1993). Also relocates review requirements related to the Emergency Plan and the Security Plan from the TS to the respective plans.

Date of Issuance: August 20, 1996

Effective Date: August 20, 1996

Amendment Nos.: 147 and 86

Facility Operating License Nos. DPR-67 and NPF-16: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49938) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 20, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

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

Date of application for amendments: June 17, 1996

Brief description of amendments: The amendments revise Technical Specification 5.3.1, Fuel Assemblies, to remove the restriction on the number of fuel rods clad with ZIRLO™ that can be loaded into the core.

Date of issuance: August 19, 1996

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

Amendment Nos.: 94, 72

Facility Operating License Nos. NPF-68 and NPF-81: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37299)

The Commission’s related evaluation of the amendments is contained in a Safety Evaluation dated August 19, 1996. No significant hazards consideration comments received: No

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

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

Date of application for amendments: June 17, 1996

Brief description of amendments: The amendments revise Technical Specification 3/4.8.1, A.C. Sources, and its associated Bases, by changing Surveillance Requirement 4.8.1.1.2.j(2) to limit the 10-year pressure test of certain portions of the diesel fuel oil system to the isolable portions of the fuel oil piping.

Date of issuance: August 28, 1996

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

Amendment Nos.: 95 and 73

Facility Operating License Nos. NPF-68 and NPF-81: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 17, 1996 (61 FR 37300) The Commission’s related evaluation of the amendments is contained in a Safety Evaluation dated August 28, 1996. No significant hazards consideration comments received: No

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

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: March 15, 1996, as supplemented July 18, 1996.

Brief description of amendment: The amendment revised TS 4.6.2.1 ‘‘Containment Systems - Depressurization Systems - Suppression Pool’’ to extend the time interval for performing the containment drywell-to-suppression chamber bypass leakage tests consistent with schedules for containment integrated leak rate testing under Option B to 10 CFR Part 50, Appendix J.

Date of issuance: August 27, 1996

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

Amendment No.: 75

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20851) The Commission’s related evaluation of the amendment is contained in a Safety Evaluation dated August 27, 1996 No significant hazards consideration comments received: No

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

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: March 20, 1996

Brief description of amendment: The amendment revises Technical Specification 3/4.3.1 ‘‘Reactor Protection System Instrumentation’’ to modify operability requirements for the Average Power Range Monitor for operational conditions 3, 4, and 5.

Date of issuance: August 28, 1996

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

Amendment No.: 76

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: May 8, 1996 (61 FR 20852) The Commission’s related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 1996 No significant hazards consideration comments received: No

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

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

Date of application for amendment: April 25, 1996

Brief description of amendment: The amendment modifies the calibration requirement for the source range monitors and intermediate range monitors by noting that the sensors are excluded.

Date of issuance: August 19, 1996

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

Amendment No.: 96

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 19, 1996 (61 FR 31183) The Commission’s related evaluation of the amendment is contained in a Safety Evaluation dated August 19, 1996. No significant hazards consideration comments received: No.

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

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

Date of application for amendments: November 14, 1994, as supplemented by letters dated December 7, 1995, February 2, 1996, May 28, 1996, and July 30, 1996.

Brief description of amendments: The amendment revised the combined Technical Specifications (TS) for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, for the slave relay test frequency from quarterly (Q) to refueling (R). The request also removed table notation 4 from Table 4.3-2. The associated Bases were revised.

Date of issuance: August 19, 1996

Effective date: August 19, 1996, to be implemented within 30 days of date of issuance.

Amendment Nos.: Unit 1 - 115; Unit 2 - 113

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62495). The supplemental letters provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission’s related evaluation of the amendments is contained in a Safety Evaluation dated August 19, 1996. No significant hazards consideration comments received: No.

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

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: June 13, 1996, as supplemented by letter dated August 7, 1996.

Brief description of amendment: This amendment will permit a one time performance of TS surveillance requirement 3.3.1.1.12 for the Average Power Range Monitor Flow Biased High Scram function with a delayed entry into associated TS Conditions and Required Actions for up to six hours provided core flow is maintained at or above eighty-two percent. This change is in effect until the end of refueling outage 2R11.

Date of issuance: August 16, 1996

Effective date: Unit 2, as of the date of issuance, to be implemented within 30 days.

Amendment No.: 216

Facility Operating License No. DPR-44: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 3, 1996 (61 FR 34895) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 16, 1996. No significant hazards consideration comments received: No

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of applications for amendment: June 15, September 15, October 25, and November 30, 1995.

Brief description of amendment: The amendments change the Technical Specifications regarding the Control Rod System, the Auxiliary Electrical Systems, the Containment Systems and the Standby Liquid Control System to reflect changes to the length of the operating cycle of 24 months.

Date of issuance: August 16, 1996

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

Amendment No.: 232

Facility Operating License No. DPR-59: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 13, 1995 (60 FR 47623), January 22, 1996 (61 FR 1633, 61 FR 1634, 61 FR 1635) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 16, 1996. No significant hazards consideration comments received: No

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

Dated at Rockville, Maryland, this 4th day of September 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

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

[Doc. 96-23032 Filed 9-10-96; 8:45 am]

BILLING CODE 7590-01-F

Draft Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a proposed revision of a guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide is a proposed Revision 2 to Regulatory Guide 1.160, and it is temporarily identified as DG–1051, ''Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.'' The guide will be in Division 1, ''Power Reactors.'' This regulatory guide is being revised to endorse Revision 2 of NUMARC 93–01, ''Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants'' (April 1996), which is an update of a Nuclear Energy Institute document. This regulatory guide will provide current guidance on methods acceptable to the NRC staff for structuring a maintenance program in accordance with the safety significance of the structures, systems, and components covered by the maintenance rule, 10 CFR 50.65.

The draft guide has not received complete staff review and does not represent an official NRC staff position.

Public comments are being solicited on the guide. Comments should be accompanied by supporting data. Written comments may be submitted to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Comments will be most helpful if received by November 15, 1996.

Although a time limit is given for comments on this draft guide, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Comments may be submitted electronically, in either ASCII text or Wordperfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FedWorld. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet.

If using a personal computer and modem, the NRC subsystem on FedWorld can be accessed directly by dialing the toll free number: 1–800–303–9672. Communication software parameters should be set as follows: parity to none, data bits to 8, and stop bits to 1 (N,8,1). Using ANSI or VT–100 terminal emulation, the NRC NUREGs and RegGuides for Comment subsystem can then be accessed by selecting the ''Rules Menu'' option from the ''NRC Main Menu.'' For further information about options available for NRC at FedWorld, consult the ''Help/Information Center'' from the ''NRC Main Menu.'' Users will find the ''FedWorld Online User's Guides'' particularly helpful. Many NRC subsystems and data bases also have a ''Help/Information Center'' option that is tailored to the particular subsystem.

The NRC subsystem on FedWorld can also be accessed by a direct dial phone number for the main FedWorld BBS, 703–321–3339, or by using Telnet via Internet, fedworld.gov. If using 703–321–3339 to contact FedWorld, the NRC subsystem will be accessed from the main FedWorld menu by selecting the ''Regulatory, Government Administration and State Systems,'' then selecting ''Regulatory Information Mall.'' At that point, a menu will be displayed that has an option ''U.S. Nuclear Regulatory Commission'' that will take you to the NRC Online main menu. The NRC Online area also can be accessed directly by typing ''/go nrc'' at a FedWorld command line. If you access NRC from FedWorld's main menu, you