Broad Street, Walhalla, South Carolina for the Oconee Nuclear Station; the York County Library, 138 East Black Street, Rock Hill, South Carolina 29730 for the Catawba Nuclear Station; and the J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223 for the McGuire Nuclear Station.

Dated at Rockville, Maryland, this 3rd day of April 1997.

For The Nuclear Regulatory Commission.

Herbert N. Berkow,

Director, Project Directorate II–2, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 97–9059 Filed 4–8–97; 8:45 am]

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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 March 17, 1997 through March 28, 1997. The last biweekly notice was published on March 26, 1997 (62 FR 14457).

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 May 9, 1997, 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.

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: March 5, 1997.

Description of amendments request: The proposed amendments would incorporate a new Technical Specification (TS) for instrumentation associated with automatic isolation of a pathway for release of non-condensible gases from the main condenser. At power levels of 5 percent or less, mechanical vacuum pumps are used to remove non-condensible gases from the condenser using a pathway to the release stack that bypasses the normal holdup and filter train. The proposed TS will require that four channels of the main steam line radiation—high isolation function be capable of tripping the mechanical vacuum pumps and closing an isolation valve in the release pathway. Surveillance requirements are included in the TS to ensure the isolation instrumentation will perform its intended function.

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

The proposed change incorporates a new Technical Specification 3/4.3.8, ''Condenser Vacuum Pump Isolation Instrumentation.'' This specification will require that the main steam line radiation—high isolation function be capable of tripping the condenser vacuum pump(s) and isolate the associated common isolation valve. Four instrumentation channels of this function are required to be operable when the unit is in OPERATIONAL CONDITION 1 or 2 with a condenser vacuum pump in operation. Adding the requirement to trip the condenser vacuum pumps does not affect the probability of an accident previously evaluated. The probability of component failure of the proposed design for condenser vacuum pump isolation devices is the same as that of the original licensing basis. As a result, the capability to isolate the condenser vacuum pump will not be significantly impacted.

CP&L contracted Scientech-NUS to recalculate the main control room doses resulting from a control rod drop accident assuming main steam line radiation monitors isolate the condenser vacuum pump(s) and determined the dose to be 23.2 rem thyroid and 0.05 rem whole body, which is less than the General Design Criterion (GDC) 19/ Standard Review Plan (SRP) Section 6.4 limits of 30 rem thyroid and 5 rem whole body. The offsite doses at the exclusion area boundary after 2 hours are 0.16 rem thyroid and 0.015 rem whole body, which is less than the SRP Section 15.4.9 limits. The low population zone (LPZ) dose is estimated to be about 1 rem thyroid, which is also well below regulatory limits. Therefore, the proposed [amendments do] not increase the consequences of an accident previously evaluated.

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

The proposed [amendments add] new requirements to ensure the capability to trip the condenser vacuum pump(s). The proposed [changes do] not affect the operability of equipment designed to mitigate the consequences of an accident nor [do they] create a potential to initiate a new type of accident. Therefore, the proposed [changes do] not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license [amendments do] not involve a significant reduction in a margin of safety.

The safety-related main steam line radiation monitors provide a highly reliable means to detect radioactivity resulting from a control rod drop accident and will provide automatic trip of the condenser vacuum pumps and isolation of the associated isolation valve. Use of the main steam line radiation monitors for this application is consistent with the original Brunswick Steam Electric Plant design for condenser pump and associated valve isolation. CP&L contracted Scientech-NUS to recalculate the main control room doses resulting from a control rod drop accident assuming main steam line radiation monitors isolate the condenser vacuum pump(s) and determined it to be 23.2 rem thyroid and 0.05 rem whole body, which is less than the GDC 19/SRP Section 6.4 limits of 30 rem thyroid and 5 rem whole body. The offsite doses at the exclusion area boundary after 2 hours are 0.16 rem thyroid and 0.015 rem whole body, which is less than the SRP Section 15.4.9 limits. LPZ dose is estimated to be about 1 rem thyroid, which is also well below regulatory limits. Therefore, the proposed [changes do] not involve a significant reduction in a margin of safety.

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

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

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

NRC Project Director: Mark Reinhart, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 18, 1997.

Description of amendment request: The proposed change revises the Plant System Turbine Cycle Technical Specification (TS) 3/4.7.1 by revising the power range high neutron flux setpoint values in TS Table 3.7–1.

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

The high flux setpoints are being revised to provide additional margin against secondary side overpressurization for LOL/TT [loss-of-load/turbine trip] events. The proposed revision will not create any loss or reduction in redundancy or diversity in the reactor protection systems that would increase the probability of a previously evaluated accident. The high flux setpoints are being revised to ensure that the consequences of a previously evaluated accident do not increase.

Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

No new or previously unanticipated failure mechanisms are introduced by the proposed change. No new failure modes have been created by the proposed change. No new credible event or initiating factor is introduced. Reactor power is limited to ensure that the secondary system is not overpressurized.

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 amendment does not involve a significant reduction in the margin of safety.

The margin of safety as defined in the basis of the Technical Specification does not decrease. This change is proposed to ensure that the secondary system pressure will be limited to within 110% of its design pressure during the most severe anticipated operational transient. The revised high flux setpoints are intended to bound the allowable operating configurations of TS Table 3.7–1.

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

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

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Mark Reinhart, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 21, 1997.

Description of amendment request: The proposed change adds a definitive time limit to Technical Specification 3.3.2 in Action 16 of Table 3.3–3 to place an inoperable channel into bypass.

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

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

The proposed change does not affect the operation or design of the plant in any way. The requirement to place the channel into bypass already exists and this change simply provides a specific time limit. This logic circuit is not an initiator of any event and with no change in logic or operation there is no change in consequences.

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

The proposed specific time limit does not involve any physical alterations or additions to plant equipment or alter the manner in which any safety-related system performs its function. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change replaces an indeterminate time period with a specific limit of six hours. Six hours is a reasonable period in which to complete this requirement and is identical to the time allowed for these functions in NUREG–1431 [Standard Specifications Westinghouse Plants]. 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 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Mark Reinhart, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 27, 1997.

Description of amendment request: The proposed change adds sleeve installation as an alternative to tube plugging for repairing degraded steam generator tubes to Technical Specification 3/4.4.5, Steam Generators.

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

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

The only equipment affected by sleeving is the steam generator tubes. The most severe malfunction of a steam generator tube is a tube rupture. The consequences of a ruptured sleeve are no greater than the consequences of a ruptured tube. Sleeving does not increase the probability of a steam generator tube failure because the sleeved tube has been shown to have a significant safety factor for burst and collapse pressures as well as demonstrated acceptable resistance to corrosion and fatigue loading. Thus, a steam generator with sleeved tubes would perform in the same manner as one without sleeved tubes.

A sleeved tube is functionally equivalent to an unsleeved tube except for less effective heat transfer due to the air gap and a slightly higher pressure drop due to the primary flow restriction. These differences are bounded by the current tube plugging limits.

Analysis and testing have demonstrated that the sleeves are structurally adequate to withstand the load existing within the steam generator tubes whether the original tube is still intact or is breeched.

There is no increase in the possibility for increased fatigue loadings. There is no possibility for the sleeve to become dislodged from its plugging location and enter the RCS [Reactor Coolant System] flow path.

The plant safety analysis for tube plugging bounds tube sleeving.

The proposed change has no significant effect on the configuration of the plant. The proposed change does not affect the way in which the plant is operated. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

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

A sleeved tube is functionally equivalent to an unsleeved tube except for less effective heat transfer due to the air gap and a slightly higher pressure drop due to the primary flow restriction. These differences are bounded by the current tube plugging limits.

The sleeved tube has been shown to have a significant safety factor for burst and collapse pressures as well as demonstrated acceptable resistance to corrosion and fatigue loading. Thus, a steam generator with sleeved tubes would perform in the same manner as one without sleeved tubes.

The proposed change has no significant effect on the configuration of the plant. The proposed change does not affect the way in which the plant is operated. 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 amendment does not involve a significant reduction in the margin of safety.

The proposed revision to permit the installation of tube sleeves does not reduce the margin of safety as presently defined in Technical Specification BASES section 3/4.4.5. This margin of safety includes primary to secondary leakage limits and tube plugging limits which are not changed by the proposed amendment. The analyses and testing of the proposed sleeve design demonstrates that the structural integrity of the RCS is maintained. Design of the tube sleeve considers mechanical/structural aspects, water chemistry and metallurgical aspects as well as thermal/hydraulic considerations.

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

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

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Mark Reinhart, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: March 10, 1997.

Description of amendment request: The proposed changes to Technical Specification 3.5.1 provide an optional method of meeting surveillance requirements by allowing the use of instrument readings to meet surveillance 4.5.1.1.a.1, and adds a new Action c to cover a condition in which one accumulator has a boron concentration not within limits.

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

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

The accumulators are not initiators of any event and so the probability of occurrence of an event is unaffected by either of the proposed changes. The use of actual instrumentation readings to comply with the surveillance does not change the function or performance of the accumulators and thus does not affect any accident consequences. The increase in the allowed time to restore the boron concentration to within limits is consistent with allowed out of service times for other Emergency Safeguards equipment.

It will not have a significant impact on subcriticality during reflood. Therefore, there will be no increase in the consequences of an accident.

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

The proposed changes to the accumulator specification do not involve any physical alterations or additions to plant equipment or alter the manner in which any safety-related system performs its function. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to the surveillance requirement provides an equivalent means of meeting the requirement. Since there is no change in either the accumulator limits or the surveillance frequency, there is no reduction in safety margin. The new Action c to address returning the boron concentration of a single accumulator to within limits allows an out of service time commensurate with the times allowed for other Engineered Safeguards Features. The boron concentration of one accumulator does not have a significant impact on subcriticality during reflood and thus does not involve a 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: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Mark Reinhart, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: March 14, 1997.

Description of amendment request: The amendment will revise the Final Safety Analysis Report to include the evaluation of a spent fuel cask drop analysis.

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 changes described do not impact the probability of occurrence of accidents previously analyzed. Removal of the valve box covers and all but four of the cask closure head sleeve nuts has no impact on accident initiators. Dose assessments using maximum potential releases assuming failure of the spent fuel and radionuclide release through the gap between the cask closure head and the cask or damage to the valves show that no significant increase in consequences of an accident previously evaluated would occur. [Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.]

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

Compromising the integrity of the cask by removing the valve box covers and closure head sleeve nuts in preparation for unloading the spent fuel from the cask does not create the possibility of a new type of accident or equipment malfunction. No safety-related equipment, safety function, or operations of plant equipment will be altered as a result of this change. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The NRC basis for acceptance of a spent fuel cask drop is documented in Section 15.7.5 of the Safety Evaluation Report, NUREG–1038, dated November 1983. It states, ''* * * no loss of cask integrity is postulated to occur in the event of a drop, and the staff concludes there will be no significant radiation released to the environment. The radiological consequences will be less than a small fraction of the 10 CFR 100 exposure guideline values.''

As described in the proposed change, even though complete cask integrity may not be preserved in the event of a loaded cask drop with the valve box covers removed or with only four, rather than 32, closure head sleeve nuts installed, the radiological consequences calculated using conservative assumptions were determined to be a small fraction of the 10 CFR 100 values. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

NRC Project Director: Mark Reinhart, Acting.

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

Date of amendment request: June 20, 1996, as supplemented by letters dated December 30, 1996, and March 5, 1997.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) by incorporating NRC approved thermal limit licensing methodology in the list of approved methodologies used in establishing the fuel cycle specific thermal limits. In addition, the proposed amendment would correct mirror editorial items in the 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, which is presented below:

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits will be established consistent with NRC approved methods to ensure that fuel performance during normal, transient and accident conditions is acceptable. The proposed Technical Specifications amendment reflects NRC approved SPC methodology used to analyze normal operations, including anticipated operational occurrences (AOOs), and to determine the potential consequences of accidents.

Licensing Methods and Models

The proposed amendment is to support operation with NRC approved fuel and licensing methods supplied from Siemens Power Corporation [SPC]. In accordance with [Updated Final Safety Analysis Report] UFSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods. The latest NRC approved revision to the Siemens [loss-of-coolant accident] LOCA analysis methodology (Reference: ANF–91–048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model) will be used to evaluate the ATRIUM–9B and other co-resident fuel types. The other licensing analysis methods and models are also NRC approved. The approved methods and models are used to determine the fuel thermal limits (e.g., average planar linear heat generation rate, transient linear heat generation rate, minimum critical power ratio and linear heat generation rate). The SPC core monitoring code enables the site to monitor k_{eff} as well as control rod density to perform the reactivity anomaly surveillance. Therefore, the change in licensing analysis methods and models does not significantly increase the probability of an accident or the consequences of an accident previously identified. The support systems for minimizing the consequences of transients and accidents are not affected by the proposed amendment.

New Fuel Design

The use of reload quantities of ATRIUM–9B fuel at Dresden does not involve a significant increase in the probability or consequences of any accident previously evaluated in the [Final Safety Analysis Report] FSAR. The ATRIUM–9B fuel is generically approved for use as a reload BWR fuel type (Reference: ANF–89–014(P)(A) Revision 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9–IX and 9X9–9X BWR Reload Fuel). Limiting postulated occurrences and normal operation have been analyzed using NRC-approved methods for the ATRIUM–9B fuel design to ensure that safety limits are protected and that acceptable transient and accident performance is maintained.

The reload fuel has no adverse impact on the performance of in-core neutron flux instrumentation or CRD response. The ATRIUM–9B fuel design will not adversely affect performance of neutron instrumentation nor will it adversely affect the movement of control blades relative to the current Dresden fuel type, the Siemens manufactured 9x9–2. The exterior dimensions of the ATRIUM–9B fuel have been evaluated by ComEd; the ATRIUM–9B fuel design provides adequate clearances relative to the co-resident 9x9–2 fuel. Thus, no increased interactions with the adjacent control blade or nuclear instrumentation are created. Additionally, given the above mentioned overall envelope similarities, no problems are anticipated with other station equipment such as the fuel storage racks, the new fuel inspection stand and the spent fuel storage pool fuel preparation machine. Therefore, the probability of adverse interactions between the ATRIUM–9B fuel and components in the core and fuel handling equipment is not significantly increased.

The ATRIUM–9B design is neutronically compatible with the existing fuel types and core components in the Dresden core. SPC tests have demonstrated that the ATRIUM–9B fuel design is hydraulically compatible with the co-resident 9x9–2 fuel. The bundle pressure drop characteristics of the ATRIUM

9B bundle are similar to those of the 9x9–2 fuel design, hence core thermal-hydraulic stability characteristics are not adversely affected by the ATRIUM–9B design. Cycle stability calculations are performed by SPC. Therefore, the probability of thermal hydraulic instability is not significantly increased.

Evaluations of the Dresden Emergency Procedures and UFSAR Chapter 15 AOOs are being performed to ensure that the use of the ATRIUM–9B fuel at Dresden does not alter any assumptions previously made in evaluating the radiological consequences of an accident at Dresden Units 2 and 3. Therefore, the radiological consequences of accidents are not significantly increased.

Methods approved by the NRC are being used in the evaluation of fuel performance during normal and abnormal operating conditions. The ComEd and SPC methods to be used for the cycle specific transient analyses have been previously NRC approved. The proposed methodologies are administrative in nature and do not significantly affect any accident precursors or accident results; as such, the proposed change to the listing of the SPC methodologies for Dresden does not significantly increase the probability or consequences of any previously evaluated accidents.

The description of the fuel is modified to include the water box design of the NRC approved ATRIUM–9B fuel type.

Review of the above concludes that the probability of occurrence and the consequences of an accident previously evaluated in the safety analysis report have not been significantly increased.

* * * * *

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation.

Licensing Methods and Models

The proposed Technical Specification amendment reflects previously approved SPC methodology used to analyze normal operations, including AOOs, and to determine the potential consequences of accidents. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and method as have been previously performed. As stated above, the proposed changes do not permit modes of reactor operation which differ from those currently permitted; therefore, the possibility of a new or different kind of accident is not created. Plant support equipment is not affected by the proposed changes; therefore, no new failure modes are created.

New Fuel Design

The basic design concept of a 9x9 fuel pin array with an internal water box has been used in various lead assembly programs and in reload quantities in Europe since 1986. WNP–2 has loaded reload quantities since 1991. Eight lead ATRIUM–9B assemblies were loaded into Dresden 2 during Cycle 15. Approximately 650 water box assemblies have been irradiated in the United States through 1995, with a substantially higher number being irradiated overseas. The NRC has reviewed and approved the ATRIUM–9B fuel design (Reference: ANF–89–014(P)(A) Revision 1 Supplement 1, Generic Mechanical Design for Advanced Nuclear Fuels 9X9–IX and 9X9–9X BWR Reload Fuel). The similarities in fuel design and operation between the ATRIUM–9B and the 9x9–2, and the previous Boiling Water Reactor experience with Siemens fuel, indicate there would be no new or different types of accidents for Dresden than have been considered for the existing fuel. Therefore, the use of ATRIUM–9B fuel at Dresden does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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3. Involve a significant reduction in the margin of safety for the following reasons:

The existing margin to safety is provided by the existing acceptance criteria (e.g., 10 CFR 50.46 limits). The proposed Technical Specification amendment reflects previously approved SPC methodology used to demonstrate that the existing acceptance criteria are satisfied. The revised LOCA methodology has been previously reviewed and approved by the USNRC for application to reload cores of BWRs. References for the Licensing Topical Reports which document this methodology, and include the Safety Evaluation Reports prepared by the USNRC, are added to the Reference section of the Technical Specifications as part of this amendment.

Licensing Methods and Models

The proposed amendment does not involve changes to the existing operability criteria. NRC approved methods and established limits (implemented in the COLR) ensure acceptable margin is maintained. The ComEd and SPC reload methodologies for the ATRIUM–9B reload design are consistent with the Technical Specification Bases. The Limiting Conditions for Operation are taken into consideration while performing the cycle specific and generic reload safety analyses. USNRC approved methods are listed in Specification 6.9.A of the Technical Specifications.

Analyses performed with USNRC-approved methodology have demonstrated that fuel design and licensing criteria will be met during normal and abnormal operating conditions. The same margins of safety will continue to be utilized by SPC (e.g., limits on peak cladding temperature, cladding oxidation, plastic strain). Therefore, there is not a significant reduction in the margin of safety.

New Fuel Design

The exterior dimensions of the ATRIUM–9B fuel assembly result in equivalent clearances relative to the co-resident 9x9–2 fuel. Thus, no increased interactions with the adjacent control blade and nuclear instrumentation are created. The change does not adversely impact equipment important to safety; therefore the margin of safety is not significantly reduced.

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 requested amendments involve no significant hazards consideration.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: March 18, 1997.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) by increasing the High Pressure Coolant Injection (HPCI) isolation setpoint from greater than/equal to 80 psig to greater than/equal to 100 psig. The licensee has requested the change to ensure consistency between the Updated Final Safety Analysis Report (UFSAR), design basis documents and the TS. The function of the setpoint is to assure the HPCI turbine steam supply is isolated in the event that the reactor scram supply pressure falls below the stall pressure of the HPCI turbine and the system seals are no longer effective in controlling the release of potentially contaminated steam.

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

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

The Low Reactor Pressure isolation of the HPCI steam supply lines is provided to prevent damage to the HPCI turbine when the reactor steam pressure has decreased below that required to provide adequate motive force to operate the system. The steam supply isolation low reactor pressure setpoint is not an assumed initiator or contributor to any previously evaluated accident and therefore this change does not involve an increase in the probability of an accident previously evaluated at Dresden Station.

The Lower Reactor Pressure isolation of the HPCI steam supply lines is described in the plant safety analysis as a backup protection to other system and facility design features which provide assurance that accident transients will not result in failures of the system which contribute significantly to the consequences of the initiating accident. The low reactor pressure isolation signal provides backup to other isolation signals to ensure isolation will occur, minimizing the radiation dose as a result of steam leakage past the turbine seals in the event of a locked rotor due to damage from liquid carryover due to postulated swell in the reactor vessel.

These analyses assume the isolation function occurs at 100 psig, and the proposed setpoint of greater than or equal to 100 psig is consistent and conservative with respect to these assumptions. Because the isolation function is not an accident initiator and the revised setpoint ensures that the isolation function continues to minimize radiological consequences, the consequences of any accident previously evaluated is not increased by the proposed changes.

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

The proposed change administratively increases the Low Reactor Vessel Pressure trip setpoint which initiates HPCI isolation. This change does not result in any new or different modes of operation. The proposed change increases the setpoint at which the HPCI turbine steam supply will be isolated as the reactor vessel pressure decreases following a postulated accident. The proposed new setpoint is conservative with respect to the existing TS limit, i.e. the new limit of greater than or equal to 100 psig is consistent and permitted by the existing limit of greater than or equal to 80 psig. The change assures that the Trip Setpoint in the TS accurately reflects the design basis and UFSAR described limits.

Because the proposed change does not result in any new modes of plant operation and administratively increases the system isolation setpoint in a conservative manner, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

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

The Trip Setpoint provides assurance that the HPCI turbine cannot be operated with a steam supply pressure too low to drive the turbine and pump. The isolation assures that the turbine does not stall and minimizes the potential for the release of radioactivity which results from steam leakage past the turbine seals. The proposed change increases the setpoint, ensuring that the required isolation occurs at a higher pressure which is more conservative, i.e. it assures the turbine is isolated before the inlet steam pressure falls to the stall pressure of the HPCI turbine and leakage occurs. The greater than or equal to 100 psig limit is well below the range of reactor vessel pressure for which HPCI is required to perform its safety function. Therefore, the margin of safety provided by the function of the HPCI isolation on low reactor vessel pressure is increased by the proposed TS change, and this change will not involve a reduction in the margin of safety.

As described, the proposed amendment for Dresden will not reduce the availability of systems required to mitigate accident conditions. Neither are new or significantly different modes of operation proposed. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

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

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

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

Date of amendment request: December 27, 1995, as supplemented September 4, October 18, and November 26, 1996.

Description of amendment request: The proposed amendment would revise technical specifications (TS) related to electrical power systems. The proposed changes include revisions to limiting conditions for operation (LCO), LCO applicability and action statements, allowed outage times (AOT), surveillance requirements (SR), and administrative controls. The changes add new requirements, revise or delete existing requirements, relocate certain existing requirements to other licensee controlled documents, and editorially restructure the proposed requirements to closely emulate the electrical power system requirements of NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants,'' (STS). The proposed requirements differ from the requirements of the STS where necessary to reflect features unique to the Palisades design. Each proposed change has been classified by the licensee as Administrative, Relocated, More Restrictive, or Less Restrictive.

Basis for proposed no significant hazards consideration determination: A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; or (3) involve a significant reduction in a margin of safety. 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:

Evaluation of ADMINISTRATIVE, RELOCATED, and MORE RESTRICTIVE changes:

ADMINISTRATIVE changes and RELOCATED changes move requirements, either within the TS or to documents controlled under 10 CFR 50.59, or [clarify] existing TS requirements, without affecting their technical content. Since ADMINISTRATIVE and RELOCATED changes do not alter the technical content of any requirements, they cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

MORE RESTRICTIVE changes only add new requirements, or revise existing requirements to result in additional operational restrictions. Since the TS, with all MORE RESTRICTIVE changes incorporated, will still contain all of the requirements which existed prior to the changes; MORE RESTRICTIVE changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

Evaluation of LESS RESTRICTIVE changes:

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

Change 3 revised the requirement for operable AC sources, using more general wording than the existing TS. The existing LCO requires that two explicitly specified transformers be operable; the proposed LCO requires that two qualified offsite circuits be operable. The proposed LCO will allow substitution of Safeguards Transformer 1–1 for Station Power Transformer 1–2 as a required AC source, but the quantity and quality of required offsite AC sources is unaffected. Since the capability and qualification of Safeguards Transformer 1–1 are equivalent to those of the Station Power transformer, neither the probability or consequences of an accident previously evaluated will be increased.

Change 10 is less restrictive only in its allowance of a 72 hour AOT for an inoperable offsite source instead of the 24 hour AOT currently required. The change also makes a considerably more restrictive change by eliminating the allowance, based on submittal of a report, for continuous operation with Startup Transformer 1–2 inoperable. Changing an AOT, alone, cannot increase the probability or consequences of an accident previously evaluated.

Change 14 allows, for an inoperable DG [diesel generator], verification that no common cause failure is involved in lieu of test starting the other DG. The intent of the test starting requirement is to verify that there is no common cause failure which also makes the other DG inoperable. The proposed action statement thereby accomplishes the same objective as that it replaces. Since the proposed action statement accomplishes the same objective as the one it replaces, operation in accordance with the proposed change will not increase the probability or consequences of an accident previously evaluated.

Change 21 revises the SR for the DG starting test. [The ''Less Restrictive'' elements of the change eliminate the requirement to vary use of the A and B starting circuits for each monthly test, because the DG is not assumed to be single failure proof; and eliminate requirements that the DGs be manually started and that they be synchronized from the control room, because no practical alternatives exist for accomplishing these actions]. The proposed change does not alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, operation of the facility in accordance with the proposed change will not involve an increase in the probability of an accident. Change 21 requires more rigorous testing of the DGs than required by the existing Technical Specifications. The more rigorous testing is intended to provide additional assurance that the DGs are capable of performing their design function and should, therefore, involve a reduction, rather than an increase, in the consequences of those accidents previously evaluated.

Change 25 revises the SR for testing the fuel transfer system. The proposed change does not alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, operation of the facility in accordance with the proposed change will not involve an increase in the probability of an accident. The only ''Less Restrictive'' feature of proposed SR is test interval extension from ''each month'' to ''each 92 days.'' Changing a surveillance frequency, alone, cannot increase the probability or consequences of an accident previously evaluated.

Change 26 revises the station battery SRs. The proposed monthly and quarterly battery SRs contain all of the test requirements of the existing SRs with two exceptions: (1) The proposed interval for measuring each cell voltage is ''each 92 days'' instead of the existing ''every month'' and (2) the requirement to record the amount of water added has been deleted. Changing a surveillance frequency or deleting a maintenance record cannot increase the probability or consequences of an accident previously evaluated.

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

Change 3 only involves the specified offsite power sources. Since the Loss of Offsite Power is already considered in the accident analyses, operating the facility in accordance with Change 3 will not create the possibility of a new or different kind of accident from any previously evaluated.

Change 10 revises an AOT; Change 14 revises a required action; Change 21 revises a testing requirement; Changes 25 and 26 revise a surveillance interval; and Change 26 deletes the requirement for a maintenance record. None of these proposed changes alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, operation of the facility in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

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

Change 3 does not alter the quantity or quality of offsite sources required to be available. Therefore, operating the facility in accordance with the proposed change will not involve a reduction in a margin of safety.

Change 10 revises an AOT; Change 14 revises a required action, Change 21 revises a testing requirement; Changes 25 and 26 revise a surveillance interval; and Change 26 deletes the requirement for a maintenance record. These proposed changes do not alter any plant operating conditions, operating practices, equipment settings, or equipment capabilities. Therefore, operating the facility in accordance with the proposed change will not involve a reduction in a margin of safety.

The licensee's September 4, 1996, supplement stated that three of the proposed changes contained in the supplement were not addressed in the December 27, 1995, no significant hazards analysis. The changes involved TS requirements that would be deleted. Equivalent requirements would be incorporated in the FSAR or other documents subject to the controls of 10 CFR 50.59. The licensee's analysis of the issue of no significant hazards consideration for these changes is presented below:

1. Do changes which relocate a requirement from the TS to documents which are controlled under 10 CFR 50.59 involve a significant increase in the probability or consequences of an accident previously evaluated?

10 CFR 50.59 specifically prohibits [without obtaining prior NRC review and approval] changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report ''if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased''. Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR [Final Safety Analysis Report] or to documents which are referenced by the FSAR cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do changes which relocate a requirement from the TS to documents which are controlled under 10 CFR 50.59 create the possibility of a new or different kind of accident from any previously evaluated?

10 CFR 50.59 specifically prohibits [without obtaining prior NRC review and approval] changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report ''if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created''. Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR or to documents which are referenced by the FSAR cannot create the possibility of a new or different kind of accident from any previously evaluated.

3. Do these changes which relocate a requirement from the TS to documents which are controlled under 10 CFR 50.59 involve a significant reduction in a margin of safety?

10 CFR 50.59 specifically prohibits [without obtaining prior NRC review and approval] changes to the facility as described in the safety analysis report, and to procedures described in the safety analysis report ''if the margin of safety as defined in the basis for any technical specification is reduced''. Since the conditions which limit changes performed under 50.59 are more restrictive than the conditions which define changes considered to involve a significant hazards consideration, relocation of a requirement from the TS to the FSAR or to documents which are referenced by the FSAR cannot involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analyses 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 N. Hannon.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: March 10, 1997.

Description of amendment request: The proposed amendment would modify Unit 1 Technical Specification (TS) 5.2.1 to add ZIRLO as fuel assembly material and add reference to Nuclear Regulatory Commission approved Topical Report, WCAP–12610, ''Vantage+ Fuel Assembly Reference Core Report'', to TS 6.9.1.12 for both units.

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

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

The methodologies used in the accident analyses have been modified to reflect the requirements provided in WCAP–12610, VANTAGE+ Fuel Assembly Reference Core Report. Reference to this NRC approved ZIRLO topical report has been added to Specification 6.9.1.12, for both units to ensure the analytical methods used to determine the core operating limits are consistent with those previously approved by the NRC. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel rod material does not adversely affect fuel performance or impact nuclear design methodology. Therefore, accident analysis results are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design bases.

VANTAGE 5H fuel assemblies with ZIRLO fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5H fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO fuel rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. Since the original design criteria are met, the ZIRLO fuel rods will not be an initiator for any new accident. The fuel rod material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the fuel rod integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO fuel rods.

The dose predictions in the safety analyses are not sensitive to the fuel rod material used; therefore, the radiological consequences of accidents previously evaluated in the safety analysis remain valid. A reload analysis is completed for each cycle, in accordance with NRC approved methodologies. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

VANTAGE 5H fuel assemblies with ZIRLO fuel rods satisfy the same design bases as those used for other VANTAGE 5H fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The ZIRLO fuel rod material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The use of Zircaloy-4, ZIRLO, or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC approved methodology will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC approved methods that have been approved for application to the fuel configuration.

Use of ZIRLO as fuel rod material does not change the VANTAGE 5H reload design and safety analysis limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the technical specifications. For each reload core, the fuel assemblies will be evaluated using NRC approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the UFSAR [Updated Final Safety Analysis Report] or any plant technical specification BASES.

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: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001.

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

NRC Project Director: John F. Stolz.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 17, 1996.

Description of amendment request: The proposed amendment would reflect that the name of Louisiana Power & Light Company, which is licensed to own and possess Waterford 3, has been changed to Entergy Louisiana, Inc.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change documents changing the legal name of the company. The proposed change will not affect any other obligations. The company will still own all of the same assets, they still serve the same customers, and all existing obligations and commitments will continue to be honored. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No

The administrative changes in the Operating License requirements do not involve any change in the design of the plant. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change is administrative in nature and does not reduce the level of safety imposed by any current requirement. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: October 16, 1996.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) action requirements 3.2.1 and 3.2.4 and their associated surveillance requirements to extend the allowable time for the Core Operation Limit Supervisory System (COLSS) to be out of service by monitoring for adverse trends in the linear heat rate (LHR) and departure from nucleate boiling (DNBR) limits.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change does not modify the requirement to operate within the alternate LHR and DNBR limits nor does it modify the actual LHR or DNBR limits themselves. In the case of exceeding a COLSS calculated [power operating limit] POL, Entergy agrees that corrective action should be initiated promptly to bring the LHR and DNBR within their respective limits and, in this case, a 15 minute time limit is appropriate. However, in the case of exceeding a [core protection calculator] CPC calculated operating limit following the loss of COLSS, it is clear that simply because COLSS execution was lost does not mean that the plant is operating outside the range of conditions assumed in the Chapter 15 Safety Analysis and, in this case, a 15 minute time limit is not appropriate. An increase from 2 hours to 8 hours to regain the monitoring capabilities of COLSS would not significantly increase the probability of exceeding the actual LHR or DNBR power operating limits since the increase in COLSS out-of-service time will be compensated for by monitoring for adverse trends of the important CPC calculated parameters (DNBR Margin and LHR). Further, since the proposed change will result in maintaining steady-state conditions while monitoring for adverse trends, it will be easier for the operators to detect any abnormal occurrence that has the potential to degrade either the LHR or the DNBR.

The primary consideration in extending the COLSS out of service time limit is the remote possibility of a slow, undetectable transient that degrades the LHR and/or DNBR slowly over the 8 hour period and is then followed by an [anticipated operational occurrence] AOO or an accident. The parameters normally monitored by COLSS which have the potential for degrading the LHR and DNBR if no corrective action is taken are: Reactor Coolant System (RCS) flow rate, axial and radial power distributions, core inlet temperature, core power, RCS pressure and azimuthal tilt. Of these parameters, core inlet temperature, core power, and RCS pressure are easily monitored by the plant operators using various safety-grade, Redundant Control Room indications and, therefore, changes in these parameters are readily apparent. Further, operating experience at Waterford 3 and other [Combustion Engineering] CE nuclear steam supply systems using the same reactor coolant pumps (RCPs) as Waterford has shown that measurable changes in RCP Δ Ps (which COLSS uses to calculate RCS flow) are very rare and when they do occur, involve abrupt step changes in flow which are readily apparent; hence, the probability of a slow degradation in the RCS flow rate is exceedingly small. Thus, the parameters that comparatively (although still remote) pose the most potential for a degradation in the core thermal margin when COLSS is out of service relate to the axial and radial core power distributions and the azimuthal tilt. These parameters are discussed below.

Axial xenon oscillations are a normal consequence of the Waterford 3 core design, particularly near the end of core life. As a result, Waterford 3 operations personnel are instructed, per operating procedure OP–10–001, General Plant Operations, to maintain strict control over the axial power shape in the core. Although the primary reason for axial shape control is to maintain an even fuel burnup throughout the core, it also results in maintaining the axial power shapes well within the limits assumed in the safety analysis. Typically, axial shape control practiced at Waterford 3 maintains the axial shape index (ASI) within 0.05 ASI units of the equilibrium shape index (ESI), which is normally very near 0.0.

Hypothetically, the most severe situation which could be postulated to occur, although again remote, would be if COLSS execution was lost just when the plant operators were ready to take manual action to return the ASI value to within the ESI + 0.05 control band. Since a full xenon oscillation takes approximately 26 hours, there would be about 6 hours from the time that control action would normally be taken to the time that the ASI reached its peak value (i.e., it takes one quarter cycle for the ASI to travel from its ESI value to its peak value). Since abnormal operating procedure OP–901–501, PMC or Core Operating Limit Supervisory System Inoperable, will be revised to require the CPC calculated LHR and DNBR trends to be monitored every 15 minutes (see below), any significant change in the axial shape index will be apparent through a change in these CPC calculated values. Hence, due to the attention given the axial power distribution, both when COLSS is in service as well as when COLSS is out of service it is very improbable that a change in ASI during eight hours of steady-state operation with COLSS out of service could be either undetected or lead to a condition that placed the reactor outside the range of initial conditions that were assumed in the safety analysis.

With regards to azimuthal tilt, there is very rarely any significant change in this parameter as long as all [control element assembly] CEAs are properly aligned. The only real contributor to a rapid increase in azimuthal tilt would be an inadvertent CEA drop; however, since the probability of a CEA drop is very low, the likelihood of this event occurring within the eight hour time limit is even lower. In the unlikely event that a CEA drop did occur, the Control Element Assembly Calculators (CEACs) provide a safety-grade, redundant means of alerting the operators that corrective action is necessary. Thus, the potential for a degradation in azimuthal tilt during eight hours of steady-state operation following the loss of COLSS is both highly unlikely and relatively easy to detect using instrumentation already available in the Control Room.

As previously stated, upon approval of the proposed change plant personnel will revise abnormal operating OP–901–501, PMC or Core Operating Limit Supervisory System Inoperable, to monitor for adverse trends of the CPC calculated values of LHR and DNBR. Currently, this procedure requires that the monitoring frequency for LHR and DNBR be increased to once every 15 minutes on a loss of COLSS.

Extending the time to restore the CPC calculated LHR and DNBR to within the acceptable operating range from 2 hours to 8 hours is being proposed to assure that COLSS can be restored thus decreasing the probability of an avoidable challenge to the reactor protection system (RPS) during a power reduction. It is possible that the required power reductions may exceed 25% near the end of the fuel cycle. These large power reductions result in a rapid increase in xenon concentration, changes in ASI, and a subsequent decrease in cold leg temperature (T-cold) that may be difficult to control. Accordingly, given the potential for power reductions of this magnitude, it is appropriate to extend the time allowed to restore COLSS so that a power reduction may be unnecessary.

Taken in total, the proposed changes will reduce the number of potentially unnecessary power reductions by allowing more time for COLSS to be restored along with the advantages of trend monitoring in detecting an adverse trend expeditiously. The proposed change will result in significant operational benefits while continuing to maintain a high degree of confidence that the core conditions remain well within the range of values assumed in the safety analysis.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No

The proposed change does not alter the current power operating limits nor does it involve any changes to COLSS or CPC software. There has been no physical change to plant systems, structures or components nor will the proposed change affect the ability of any of the safety-related equipment required to mitigate AOOs or accidents. The only significant change associated with the proposed amendment involves changes to the operating procedures used when COLSS is out-of-service. All revisions to operating procedures will be reviewed and approved by appropriate plant personnel as required by the Administrative Controls (Section 6) in the Waterford 3 Technical Specifications. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No

The intent of [limiting conditions for operation] LCOs 3.2.1 and 3.2.4 is to maintain the reactor within the range of initial conditions that was assumed in the Safety Analysis. Maintaining the LHR within the specified range ensures that in the event of a LOCA, the fuel cladding temperature will not exceed the 2200°F limit imposed by 10CFR46 [10 CFR Part 46]. Maintaining the DNBR within the specified range ensures that no AOO will result in a violation of the [Specified Acceptable Fuel Design Limits] SAFDLs and that no postulated accident will result in consequences more severe than those described in Chapter 15 of the [Final Safety Analysis Report] FSAR. Since there has been no change to the requirement to operate the reactor within the LHR and DNBR limits and no change to the actual LHR and DNBR limits themselves, the accident analyses described in Chapter 15 of the FSAR will not be affected and will therefore remain bounding.

The proposed change will reduce the number of potentially unnecessary power reductions along with the rate at which the power reductions are accomplished. Maintaining steady-state conditions for up to eight hours after the loss of COLSS while monitoring the CPC LHR/DNBR for trends, provides plant personnel with a reasonable period of time to return COLSS to service while continuing to maintain a high degree of confidence that the core conditions remain well within the range of values assumed in the safety analysis. In fact, monitoring for trends in LHR and DNBR Margin increases the margin of safety by allowing the anticipation of degradation in LHR or DNBR Margin. Moreover, by reducing the number of plant transients there will be an attendant reduction in probability of an AOO and subsequent RPS actuation. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn, 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: October 16, 1996.

Description of amendment request: The following changes to the Waterford Steam Electric Station, Unit 3, Technical Specifications are proposed: 1) Relocation of certain administrative controls to the Quality Assurance Program Manual (QAPM) as described in Nuclear Regulatory Commission Administrative Letter 95–06, ''Relocation of Technical Administrative Controls related to Quality Assurance''; 2) Change shift coverage from 8-hour day, 40-hour weeks to an option of 8 or 12 hour days and nominal 40-hour weeks; 3) Make certain editorial changes to the titles of certain organizational positions.

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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The conditions as they exist in the present Technical Specifications do not have an affect on either the probability or consequences of a previously evaluated accident. These changes also will have no impact to increase either the probability or consequences of a previously evaluated accident.

The proposed changes will have no affect on design basis accidents nor will the change directly affect any material condition of the plant that could directly contribute to causing or mitigating the effects of an accident.

Therefore, the proposed changes will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No

The proposed changes will not alter the operation of the plant or the manner in which it is operated. The changes do not involve a design change and do not introduce any new failure modes.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in margin of safety?

Response: No

The proposed changes are administrative in nature and affect only Section 6.0 of the Technical Specifications. The Waterford 3 margins of safety are defined in Sections 2 through 5 and are unaffected by these changes. Moving the reviews from the TS to the QAPM will have no affect on the margin of safety because reviews will still be performed. The only difference is the reviews will be administratively controlled by the QAPM. The QAPM is controlled by 10CFR50.54 so no changes can be made which would lessen these commitments (i.e., remove or reduce the requirement for procedure reviews) without prior NRC approval.

Changing from an 8 hour to an 8 or 12 hour shift will not have an adverse impact on personnel performance. The NRC study documented in NUREG CR–4248 has identified that personnel errors have decreased and productivity has increased where this change has been implemented.

Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

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

Local Public Document Room Location: University of New Orleans

Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed change modifies Technical Specification 3/4.5.2, ''ECCS Subsystems Modes 1, 2, and 3.'' The proposed change adds a surveillance requirement to verify the Emergency Core Cooling System (ECCS) piping is full of water at least once per 31 days. A change to the Technical Specification Basis 3/4.5.2 and 3/4.5.3 has been included to support this change.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change will not affect the assumptions, design parameters, or results of any accident previously evaluated. The proposed change does not add or modify any existing equipment. The proposed change adds a new surveillance requirement which will minimize the likelihood of a pressure transient occurring during system startup and provide increased assurance that the ECCS will perform its design basis function when needed. The new [low pressure safety injection] LPSI and [high pressure safety injection] HPSI vent valves which may be manipulated during this surveillance will be administratively controlled and will be locked close when not in use to prevent the possibility of a flow diversion. This surveillance requirement is consistent with NUREG 1432.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

While new vent lines are being installed under 10CFR50.59, this proposed change adds only a new surveillance requirement to Technical Specification 3/4.5.2 and therefore does not involve modifications to any existing equipment. The new vent valves, when required, will be operated and controlled in the same manner as existing LPSI and HPSI vent valves. The new LPSI and HPSI vent valves will be administratively controlled and will be locked close when not in use. This surveillance requirement is consistent with NUREG 1432.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

The functionality of ECCS is maintained such that it is capable of performing its design function as assumed in the Updated Final Safety Analysis Report. Verifying the ECCS is full of water at least once per 31 days will minimize the likelihood of a pressure transient occurring during system startup and provide increased assurance that the ECCS will perform its design basis function when needed. This surveillance requirement is consistent with NUREG 1432.

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

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: William D. Beckner.

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: March 27, 1997.

Description of amendment request: The proposed change modifies Technical Specification (TS) surveillance requirements 4.5.2.d.3 and 4.5.2.d.4. The proposed change specifies granular trisodium phosphate dodecahydrate (TSP), increases the minimum required amount of TSP that is maintained in containment during power operation, and adjusts the TSP sampling requirement accordingly. A change to the TS Basis 3/4.5.2 has been included to support this change.

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

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

Granular trisodium phosphate dodecahydrate is stored in the containment lower level to raise the pH of the sump and spray water following a LOCA. As the pH of the water increases, more radioactive iodine is kept in solution and the amount of airborne radioactive leakage is decreased. This also lessens the potential for boric acid solution reacting with galvanized metal in containment to release hydrogen. An additional advantage of a higher pH is the beneficial reduction in chloride stress corrosion cracking of metal components in the containment following an accident.

This chemical is an accident mitigator, not an accident initiator in that it is not used until after an accident has occurred. At the time it goes into solution, the accident has occurred, containment spray has been activated and water has collected in the sump. Therefore, increasing the Technical Specification minimum amount verified to be in containment or changing the sample solution and sample size will not involve a significant increase in the probability of an accident previously evaluated.

At the time TSP goes into solution, the accident has occurred, containment spray has been activated and water has collected in the containment sump. At Waterford 3, the iodine partition factor is a constant 50% and does not vary with pH as allowed in the Standard Review Plan (SRP) revision 1. The curve in SRP 6.5.2 revision 1 allows a partition factor of at least 50% for containment water at a pH of 6.5 or less. The partition factor increases as pH rises. But, the curve is based on sodium hydroxide which is much more reactive than TSP. Therefore, increasing the Technical Specification minimum amount verified to be in the containment, and corresponding sample size, will not involve any significant increase in the consequences probability of an accident because no credit is taken for reducing the amount of volatized iodine normally associated with a 7.0 pH solution.

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

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of accident from any accident previously evaluated?

Response: No.

The addition of more TSP does not represent a significant change in the configuration or operation of the plant. Trisodium phosphate dodecahydrate is currently present in the containment lower level. Design Change 3491 which increases the storage capacity of the TSP storage baskets was evaluated in accordance with 10 CFR 50.59 and found not to involve an unreviewed safety question.

Boric acid acts as a buffer to prevent the pH from rising above approximately 8.1 as TSP is dissolved. An internal study (EC–S96–013 revision 0) has shown that given the ''ratio of grams of TSP to liters of 3000 ppm boron solution'' stays less than 5.6, TSP cannot increase pH above 8.2. As pH increases, components composed of aluminum, zinc, or copper become vulnerable to corrosion. Branch Technical Position MTEB 6–1 implies that a solution pH greater than 7.5 enhances the chance for hydrogen generation as a result of aluminum corrosion. Waterford 3 administratively limits the amount of aluminum in containment to minimize the amount of hydrogen expected during a DBA. Zinc is a component of the paint applied to surfaces inside containment. The hydrogen recombiner design basis includes 464 square feet (1040 pounds) of aluminum and 419,300 square feet (17,252 pounds) of metallic zinc. Estimates of the amount of hydrogen produced by the aluminum assumes that the corrosive agent is sodium hydroxide—a much more active chemical than is TSP. Thus, the amount of hydrogen expected in the FSAR for the hydrogen recombiner bounds what would actually be produced by TSP even at a pH of approximately 8.1.

The 4.5.2.d.3 proposed TSP to boron ratio assures that pH cannot rise above 8.1 as long as post accident in-containment boric acid solution concentration is no greater than 3011 ppm boron and no less than 1504 ppm boron. The main variable in post accident concentration (the difference between 1504 and 3011) is the concentration in the RCS at the time of the accident.

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

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No.

Trisodium phosphate dodecahydrate is stored in the containment lower level to raise the pH of the sump and spray water following a LOCA. As the pH of the water increases, more radioactive iodine is kept in solution and the amount of airborne radioactive leakage is decreased. A neutral pH also reduces the hydrogen generation from the corrosion of the galvanized materials in containment. An additional advantage of a higher pH is the beneficial reduction in chloride stress corrosion cracking of metal components in the containment following an accident.

Technical Specification 4.5.2.d.3 requires verification that a minimum volume of TSP is contained in the storage baskets in containment. Nine previous runs of surveillance requirement 4.5.2.d.4 (and similar tests) showed that the TSP actually used in the plant properly neutralized a sample of water borated within RWSP boron concentration limits. Boron concentrations of eight of the sample solutions used in these tests ranged from 1753 ppm to 2217 ppm and resulted in a pH of 7.02 or greater. (The boron concentration of one test performed in 1986 was unavailable.) The ratio 4 grams to 4 liters is the amount of TSP needed to bring the solution to a pH of at least 7.0 given that the solution is in the 1753 to 2217 ppm Boron range.

The amount of TSP in containment currently is adequate assuming that RCS boric acid concentration stays below 454 ppm. However, the fuel cycle is nearly over and a restart with a refreshed core would require substantially more boric acid. We expect that the containment water would reach approximately 2400 ppm under ideal circumstances during cycle 9. During cycle 10, boron concentration in containment could reach 3011 under those same ideal conditions. As the maximum boron concentration increases, there is a non-linear increase in the amount of TSP needed to raise solution pH to 7.0. Thus, we request that the minimum amount of TSP in containment required by 4.5.2.d.3 to be increased from 97.5 cubic feet to 380 cubic feet. This change also proposes to adjust the 4.5.2.d.4 specified increase that sample solution and the TSP sample size accordingly. This change will ensure the safety injection containment sump, when filled with water, will have an acceptable pH following a LOCA. The test will not only demonstrate that TSP is in the baskets but also shows that the amount of TSP in containment can neutralize the solution expected in containment during any DBA.

Therefore, the proposed change will not involve a significant reduction in a margin of safety. The amount of iodine kept in solution during a DBA is limited to 50%. Note, the pH scale is logarithmic so that the amount of TSP needed to raise pH to 7.0 is more than three times the amount needed to reach 6.5. Furthermore, the amount of hydrogen generated during a DBA is over estimated by the analysis when it used sodium hydroxide as the corrosive agent rather than TSP.

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

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

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn, 1400 L Street N.W., Washington, D.C. 20005–3502.

NRC Project Director: William D. Beckner.

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

Date of amendment request: May 17, 1996, as supplemented March 17, 1997.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) Section 3/4.4.5, Steam Generators, 3/4.4.6, Reactor Coolant System Leakage, and associated Bases to allow the installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes. The proposed change would also specify the Westinghouse topical reports to be used for sleeve design and inspection, and identify the inspection sample size for repaired tubes. This application was previously published in the **Federal Register** on May 29, 1996, (61 FR 26938).

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:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. Listing the specific Westinghouse topical reports in the TS binds the South Texas Project (STP) to the sleeve design and inspection techniques identified in that revision of the topical report. Any changes to sleeve design or inspection technique would require a separate TS amendment.

New TS Table 4.4–3, Steam Generator Repaired Tube Inspection, identifies the inspection sample size for steam generator tubes that have already been repaired. This table simply identifies inspection criteria and associated actions for repaired tubes and does not increase 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. Implementation of laser welded sleeving maintains overall tube bundle structural and leakage integrity conditions. Providing specific Westinghouse topical report references in the TS only serves to identify which sleeve design and inspection techniques are being employed at STP. Likewise, the addition of Table 4.4–3 clarifies the expected inspection samples for previously repaired tubes. The addition of Table 4.3–3 provides assurance that previously repaired tubes will be inspected at regular intervals and appropriate action taken if the tube is found defective. Neither of these additions to the TS will create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety. Both of these changes are being added to clarify the STP steam generator tube inspection program and provide more specific detail regarding steam generator tube inspection samples and inspection techniques. By requiring inspection of previously repaired tubes, the margin of safety is increased rather than decreased.

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: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: January 28, 1997.

Description of amendment request: The proposed amendment would relocate the details of Technical Specification (TS) Section 6.2.3 on the Independent Safety Engineering Group (ISEG) from the Administration Controls section of the TSs and place these details in the Updated Final Safety Analysis Report (UFSAR) for South Texas Project, Units 1 and 2. This relocation is administrative only, and would not render any changes to the existing plant philosophy toward the ISEG or any safety analysis. Section 6.2.3 would be deleted from the TSs and removed from the table of contents for Administrative Controls. Currently UFSAR Section 13.4.2.2 describes the ISEG, but not in the detail as the current TSs.

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

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

The proposed changes move details from the Technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes do not result in any hardware or operating procedure changes. The details being removed from the Technical Specifications [TSs] are not assumed to be an initiator of any analyzed event. The UFSAR, which will contain the removed Technical Specification [TS] details, will be maintained using the provisions of 10 CFR 50.59 and is subject to the change control process in the Administrative Controls Section of the Technical Specifications [TSs]. [In addition] any changes to the UFSAR will be evaluated per 10 CFR 50.59, no increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC [Nuclear Regulatory Commission] approval. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes move details from the technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes will 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 changes move detail from the Technical Specifications [TSs] to the Updated Final Safety Analysis Report (UFSAR). The changes do not reduce the margin of safety since the relocation of details [is an administrative action and] has no impact on any safety analysis assumptions. In addition, the detail transposed from the Technical Specifications [TSs] to the UFSAR are the same as the existing Technical Specification [TS] [6.2.3]. [In addition] any future changes to the FSAR will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be allowed without prior NRC approval. [Therefore, the licensee concluded that the changes will not involve a significant reduction in a margin of safety.]

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

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

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

NRC Project Director: William D. Beckner.

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

Date of amendment request: February 7, 1997.

Description of amendment request: The proposed Technical Specification changes would clarify and/or modify instrument calibration, functional, and response time requirements for resistance temperature detector and thermocouple testing. Also, certain definitions would be clarified and/or modified using applicable wording from NRC's NUREG–1433, ''Standard Technical Specifications,'' Revision 1, and industry recommendations. Additionally, the change would relocate the reactor protection system logic response time value utilizing the guidance provided by NRC's Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' with the exception of relocating the value to the Technical Specifications Bases Section instead of the Updated Final Safety Analysis Report. The proposed amendment is intended to clarify instrumentation surveillance requirements, thereby helping to ensure proper testing of safety-related components.

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:

Pursuant to 10 CFR 50.92, NNECO [Northeast Nuclear Energy Company] has reviewed the proposed changes and concludes that the changes do not involve a significant hazards consideration (SHC) since the proposed changes satisf[y] the criteria in 10 CFR 50.92(c). That is, the proposed changes do not:

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

The proposed amendment continues to ensure the surveillance requirements satisfy the licensing basis. The current TS [technical specifications] definition for Instrument Functional Test requires injection of a simulated signal into the primary sensor to verify proper response. Current TS exempt the sensors of specific instrument channels where it is not practical to include them within the functional test boundaries. Some examples of these exemptions include neutron monitoring system, turbine control valve fast closure, and standby gas treatment initiation radiation monitors. In these cases, TS permit the performance of the functional test by injection of a simulated electrical signal into the measurement channel. The proposed definition, which is consistent with the STS [standard technical specifications] definition, for CHANNEL FUNCTIONAL TEST requires injection of the simulated signal ''as close to the sensor as practicable.'' Therefore, the proposed definition is consistent with the current TS definition and its exemptions. The primary sensor is the transmitter or switch or radiation monitor. The definition does not include sensing elements such as radiation detectors, flow elements, acceleration relays or reference legs.

This change will allow the channel functional test to be performed by means of any series of sequential, overlapping, or total channel steps and aligns this methodology with industry practice. This change does not affect accident precursors and thus does not involve a significant increase in the probability of an accident previously evaluated. The proposed change will allow a simulated or actual signal to be used to perform an Instrument or Channel Functional Test. This change does not impose a requirement to create an actual signal, nor does it eliminate any restriction on producing an actual signal. While creating an ''actual'' signal could increase the probability of an event, existing procedures (and the 10 CFR 50.59 control of revisions to them) dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations or the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the Instrument or Channel Functional Test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Because the method of initiation will not affect the acceptance criteria of the Instrument or Channel Functional Test, the change does not involve a significant increase in the consequences of an accident previously evaluated.

Minor word differences from STS are required to provide consistency with current TS wording and support the current licensing basis. These minor word differences including Industry/TSTF [Technical Specification Task Force] Standard Technical Specification Change Traveler (TSTF–64) do not alter the meaning of instrument testing in the STS or change the current licensing basis.

Moving the RPS [Reactor Protection System] Logic Response Time LCO [Limiting Condition of Operation] description to the TS definition section is an administrative change and does not alter the original intent or licensing basis.

Relocation of the RPS Logic Response Time value from the TS to the Bases section involves the use of an alternate regulatory process for controlling the instrument response time limit. The change does not introduce any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements. Any change in the RPS logic response time value would be evaluated pursuant to the requirements of 10 CFR 50.59.

The surveillance section editorial change does not alter the meaning of surveillance applicability. Providing RPS Logic Response Time surveillance frequency and applicable trip functions ensures proper testing of RPS components and is consistent with industry practice. An evaluation completed by GE [General Electric] verified the applicable RPS trip functions that require a specific logic response time using the current accident analysis as the basis. For trip functions where no explicit credit is taken in the safety analysis, the measurement of logic response time is not important, and therefore, not warranted. In addition, we have concluded, that instrumentation response time requirements (specified limits) other than RPS logic are not important to test, especially considering the long delays already accounted for in the accident analyses associated with the start of emergency power sources, ECCS [Emergency Core Cooling System] components, and containment isolations, and that the non-RPS logic response times, including response times of other instrumentation such as radiation monitors, are not part of the Millstone Unit No. 1 licensing basis. The sensors associated with all TS instrumentation are functionally tested and calibrated to ensure proper operation.

No physical change is being made to instrument channels, or to any systems or component that interfaces with the instrumentation channels, therefore there is no change in the probability or consequences of any accident analyzed in the UFSAR [Updated Final Safety Analysis Report].

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

The proposed change does not result in any design or physical configuration changes to the instrumentation channels. Operation incorporating the proposed change will not impair the instrumentation channels from performing as provided in the design basis.

Changing the TS to be consistent with current industry practice adopted in STS will help to prevent unnecessary removal and potential damage of the temperature detectors (for sensor calibration). Clarification of RPS Logic Response Time testing requirements consistent with the current licensing basis will ensure proper testing of safety-related components.

Wording changes to Instrument Calibration and Functional Test definitions do not involve a physical modification to the plant. The injection of an actual or simulated signal as close to the sensor as practical minimizes the likelihood of any transients.

Minor word differences from STS are required to provide consistency with current TS wording and support the current licensing basis. These minor word differences, including Industry/TSTF Standard Technical Specification Change Traveler (TSTF–64), do not alter the meaning of instrument testing in the STS or change the current licensing basis.

Moving the RPS Logic Response Time LCO description to the TS definition section is an administrative change and does not alter the current licensing basis.

Relocation of the RPS Logic Response Time value involves the use of an alternate process for controlling the instrument response time limits. Therefore, the above change does not introduce any accident initiators as it does not involve any new modes of plant operation, make any physical changes, alter any operational setpoints, or change the surveillance requirements.

The surveillance section editorial change does not alter the meaning of surveillance applicability. Providing RPS Logic Response Time surveillance frequency and applicable trip functions ensures proper testing of RPS components and is consistent with industry practice.

Since the proposed changes in the Technical Specifications do not adversely impact the reliability of the RPS and other automatic actuations, no new or different kind of accident is created.

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

Because the proposed change does not involve the addition or modification of plant equipment, is consistent with the existing Technical Specifications, current industry practices as outlined in NUREG 1433, ''Standard Technical Specifications GE Plants, BWR/4,'' Revision 1, and with the current design and licensing basis of the Protective Instrumentation systems including the accident analysis, no action will occur that will involve a significant reduction in a margin of safety.

The proposed change to allow the use of an actual signal in addition to the existing requirement, which limits use to a simulated signal, will not affect functional test acceptance criteria. Therefore, the proposed change does not adversely affect the reliability of the RPS or other automatic actuation and does not involve a significant reduction in a margin of safety.

Relocation of the RPS Logic Response Time value from the TS to the Bases section involves the use of an alternate regulatory process for controlling the instrument response time limit. Any change in the RPS logic response time value would be evaluated pursuant to the requirements of 10 CFR 50.59.

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 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: 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.

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment request: November 25, 1996, as supplemented December 12, 1996.

Description of amendment request: The proposed amendment would make changes to Section 2.1.A for the Safety Limit Minimum Critical Power Ratio (SLMCPR) and to Section 3.11.C for the Operating Limit Minimum Critical Power Ratio (OLMCPR). The proposed change to Section 2.1.A revises the SLMCPR value from 1.07 to 1.08 for two recirculation pump operation and from 1.08 to 1.09 for single loop operation. The proposed change to Section 3.11.C deletes the sentence that specifies the OLMCPR limit penalty for single recirculation loop operation and adds a statement that references the Core Operating Limits Report (COLR) as the source for this information.

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

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

The basis of the MCPR [minimum critical power ratio] Safety Limit calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and fuel damage in the event of a postulated accident. The probability of fuel damage is not increased. The derivation of the revised SLMCPRs for Monticello for incorporation into the Technical Specification, and its [their] use to determine cycle-specific thermal limits, have been performed using NRC-approved methods as identified in Technical Specification 6.7.A.7.b. NSP [Northern States Power] methodology established OLMCPR such that integrity of the SLMCPR is maintained for the bounding analyzed transients. Additionally, GENE [General Electric Nuclear Energy] interim implementing procedures, which incorporate cycle-specific parameters, have been used. Based on the use of these calculations, the calculation of the revised SLMCPRs maintains the integrity of the safety limits and therefore cannot increase the probability or severity of an accident. The single loop OLMCPR evaluation was performed using NSP methodology approved by the NRC. Relocating the OLMCPR value to the COLR establishes appropriate control on a core operating limit which may vary from cycle to cycle because it is cycle dependent. Since OLMCPR is developed using procedures approved in the Technical Specifications, placing the OLMCPR in the COLR cannot result in a change not controlled by the Technical Specifications. The change does not affect failure modes of equipment, therefore, this amendment will not cause a significant increase in the probability or consequences of an accident previously evaluated.

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

The MCPR Safety Limit is a Technical Specification numerical value, designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. It cannot create the possibility of any new type of accident. The new SLMCPRs have been calculated using NRC-approved methods and the OLMCPR values are more conservative. Additionally, interim procedures, which incorporate cycle-specific parameters, have been used. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident, from any accident previously evaluated.

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

The MCPR Safety Limit is a Technical Specification numerical value, designed to ensure that fuel damage from transition boiling does not occur as a result of the limiting postulated accident. Increasing the SLMCPR and OLMCPR values results in an increase in the margin of safety to fuel failure, and does not affect other plant systems. Therefore, the proposed Technical Specification change does not involve a significant reduction in the margin of safety.

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

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

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

NRC Project Director: John N. Hannon.

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

Date of amendment request: November 20, 1996, as supplemented by letter dated February 20, 1997.

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to allow the Vice President to designate the Safety Audit and Review Committee (SARC) Chairperson, to change the work hours limitation in accordance with guidance in GL 82–12, ''Nuclear Power Plant Staff Working Hours;'' to change radioactive shipments record retention requirements to comply with recent 10 CFR Part 20 changes; to revise position titles to reflect organizational changes; and other editorial changes. The February 20, 1997, supplemental letter differs from the November 20, 1996, application which was noticed in the **Federal Register** on January 2, 1997 (62 FR 131), in that the previous application did not propose changes to TS 5.3, 5.5, 5.6, 5.7, and 5.11 reflecting recent organizational changes.

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

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

The changes requested are administrative in nature. Paragraph 3.D was placed in the License by Amendment No. 155 to authorize Omaha Public Power District (OPPD) to increase the storage capacity of the FCS spent fuel pool. Amendment No. 155 stated that the TS as issued would be effective when the last new rack was installed. Since the last new rack was installed on August 8, 1994, Paragraph 3.D is no longer necessary and should be deleted from the License.

Table of Contents, Section 6.0, ''Interim Special Technical Specifications,'' Subsections 6.1 through 6.4 are proposed for deletion because all of the Specifications referred to have been deleted by previous Amendments.

The revision proposed for TS 2.15 (Item 2C of Table 2–3 & Item 1C of Table 2–4) will insert the correct terminology (Pressurizer Low/Low Pressure) into the Functional Unit description.

The revision proposed for TS 5.2 will delete the specific working hours as stated and relocate these requirements to the Updated Safety Analysis Report (USAR). Overtime will remain controlled by plant administrative procedures with the USAR generally following the guidance of the NRC's Policy Statement on working hours contained in Generic Letter 82–12, ''Nuclear Power Plant Staff Working Hours.'' Specifying personnel working hours in TS does not meet any of the four criteria contained in 10 CFR 50.36 for inclusion in the TS. Revisions to plant procedures containing these requirements are required to be evaluated in accordance with 10 CFR 50.59. The proposed relocation is similar to recent Amendments issued to the Davis-Besse Nuclear Power Station and the San Onofre Nuclear Generating Station.

The revision proposed for TS 5.5.2.2 will replace the specific title of the Chairperson of the Safety Audit and Review Committee and replace it with ''Member as appointed by the Vice President.'' This will allow the flexibility to change chairmanship of the committee amongst the members.

The revisions proposed to TS 5.3, 5.5, 5.6, 5.7, and 5.11 revise position titles and reporting responsibilities to reflect organizational changes. Qualifications for individuals in these positions meet or exceed the previous requirements.

The revision to TS 5.10 concerning retention of records of radioactive shipments will update the TS to current 10 CFR 20 requirements. Plant procedures already comply with current 10 CFR 20 record retention requirements. The addition of the Section 5.0 title corrects a minor format discrepancy.

These proposed revisions are administrative in nature. The proposed revisions have no effect on any initial assumptions or operating restrictions assumed in any accident, nor do these changes have any effect on equipment required to mitigate the consequences of an accident. Therefore the proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed revisions correct minor errors, remove outdated information, are consistent with changes in organizational structure, 10 CFR Part 20, or the criteria contained in 10 CFR 50.36. These changes will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoints or limits. No new operating modes are proposed as a result of these changes. Therefore the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The revisions listed above correct minor errors, remove outdated information, or are consistent with changes in organizational structure, 10 CFR Part 20, or the criteria contained in 10 CFR 50.36. These changes will not result in any physical alterations to the plant configuration, changes to setpoint values, or changes to the application of setpoint or limits. Therefore the proposed changes do not involve a significant reduction in a margin of safety.

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

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William H. Bateman.

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

Date of amendment request: February 26, 1997.

Description of amendment request: The proposed amendment would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 3/4.4.5 and 3.4.6.2, including associated Bases 3/4.4.5 and 3/4.4.6.2, to allow the implementation of steam generator (SG) tube voltage based repair criteria for outside diameter stress corrosion cracking (ODSCC) indications at tube-to-tube support plate (TSP) intersections. The allowed primary-to-secondary operational leakage from any one SG would be reduced from 500 gpd to 150 gpd.

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.

Structural Integrity Considerations

The structural criteria ensure that all indications subjected to voltage-based repair limits will be able to withstand pressure loading consistent with the criteria of NRC Regulatory Guide (RG) 1.121.

Tube burst criteria are inherently satisfied during normal operating conditions because of the proximity of the tube support plate (TSP). It is conservatively assumed that the entire crevice region is uncovered during the secondary side blowdown of a main steam line break (MSLB). Therefore, during a postulated MSLB accident, tube burst capability must exceed the RG 1.121 criterion requiring a margin of 1.43 times the steam line break pressure differential on tube burst.

Based on the latest industry database, the RG 1.121 criterion is satisfied by bobbin coil indications of outside diameter stress corrosion cracking (ODSCC) with signal amplitudes less than 8.7 volts. The latest NRC-approved database will be used for repair and analysis applications.

Industry testing of model boiler and operating plant tube specimens for free-span tubing (no tube support plate (TSP) restraint) at room temperature conditions show typical burst pressures in excess of 5,000 psi for ODSCC indications with voltage measurements at or below 8.7 volts. This tube burst capability exceeds the RG 1.121 criterion.

The lower voltage repair limit is conservatively defined to be 2.0 volts in accordance with NRC Generic Letter (GL) 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' August 3, 1995. This 2.0 volt repair limit is very conservative because it contains a large safety margin, based on a structural limit of 8.7 volts. A maximum allowable upper repair limit (URL) is also established using the guidance of GL 95–05. The URL is calculated before each inspection by subtracting the NDE uncertainty and growth rate allowances from the current structural limit. The URL for near term inspections at DCPP Units 1 and 2 is expected to be about 5.0 volts. Bobbin indications greater than 2.0 volts and less than or equal to 5.0 volts that are confirmed by RPC will be repaired. Bobbin indications greater than 5.0 volts will be repaired.

Following each inspection, burst probability analyses are performed for the end of cycle (EOC) distribution. In accordance with GL 95–05, the projected MSLB burst probability must be less than the threshold value of 1 x 10^{-2}. Based on the relatively small number and voltages of ODSCC indications identified to date at DCPP Units 1 and 2, it is expected that the near term EOC conditional burst probability for a faulted SG will be much less than this threshold value, providing further assurance of acceptable structural integrity.

Leakage Considerations

PG&E will implement reduced operational leakage limits as recommended in GL 95–05. PG&E will revise the TS to implement a maximum leakage rate of 150 gpd for any one SG to help preclude the potential for excessive leakage during power operation in Modes 1 and 2. The TS has also been changed to specify that the 150 gpd leak limit is not necessarily a limiting condition for operation in Modes 3 and 4. The 150 gpd leak rate per steam generator has been established for normal operation. This leakage rate provides added assurance against tube rupture at normal and faulted conditions. In Modes 3 and 4, there is less differential pressure across the tube and the potential source term from a tube failure is much less than in Modes 1 and 2. The operational leak rate monitoring program is a defense-in-depth measure that provides a means for identifying leaks during power operation to allow for repair before such leaks can result in tube failure. The leakage criteria ensure that for indications subjected to voltage-based repair criteria, induced leakage under worst-case MSLB conditions will not result in offsite and control room dose releases that exceed the applicable guideline values of 10 CFR 100 and GDC 19.

Relative to the expected leakage during accident condition loadings, a postulated MSLB outside of containment, but upstream of the main steam isolation valve (MSIV), represents the most limiting radiological condition for implementation of voltage-based repair criteria. The steam generator tubes are subjected to an increase in differential pressure following a MSLB, resulting in a postulated increase in leakage and associated offsite doses. Leakage following a MSLB bypasses containment.

PG&E will calculate the primary-to-secondary leakage for degradation subjected to the voltage repair criteria under worst-case postulated MSLB conditions. The leak rate will be compared to the maximum allowable leak rate limit of 12.8 gpm to ensure that a postulated MSLB occurring at EOC would not result in radiological consequences that are in excess of the applicable offsite and control room dose guidelines of 10 CFR 100 and GDC 19. Based on the relatively small number of ODSCC indications identified to date at DCPP Units 1 and 2, it is expected that the near term EOC predicted leak rates for a faulted SG will be much less than the maximum allowable leak rate limit.

Therefore, based on the structural integrity and leakage considerations discussed above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Implementation of the proposed voltage-based repair criteria for ODSCC at TSP intersections does not introduce any significant change to the plant design basis. Use of the criteria does not create a mechanism which could result in an accident in the free span because the repair criteria do not apply to tubes containing ODSCC located outside the thickness of the TSPs. Based on the burst probability acceptance limit of 1×10^{-2}, it is expected that for all plant conditions, neither a single nor multiple tube rupture event would likely occur in a steam generator where voltage-based repair criteria have been applied.

Steam generator tube integrity is continually maintained through inservice inspection and primary-to-secondary leakage monitoring. Any tubes with ODSCC degradation in excess of the URL are repaired.

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

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

The use of the bobbin probe to disposition ODSCC degraded tubes within TSP intersections by voltage-based repair criteria is demonstrated to maintain SG tube integrity in accordance with the requirements of RG 1.121. RG 1.121 describes a method acceptable to the NRC Staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of SG tube rupture. This is accomplished by determining the limiting conditions of degradation of SG tubing, as established by inservice inspection, for which tubes with unacceptable degradation are removed from service. Upon implementation of the voltage-based repair criteria, even under the worst case conditions, the occurrence of ODSCC at TSP intersections is not expected to lead to a SG tube rupture during normal or faulted plant conditions, nor is it expected to lead to unacceptable primary-to-secondary leakage.

In addressing the combined effects of a loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the SGs, as required by GDC 2, it has been determined that tube collapse may occur based on analysis for a large break LOCA plus SSE. The analysis identifies a maximum of 7.5 percent of tubes per SG located adjacent to wedge regions that are subject to potential collapse during combined LOCA and SSE. Tubes located in the wedge region exclusion zone will be excluded from application of voltage-based repair criteria. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced.

Implementation practices using voltage-based repair criteria bounds RG 1.83 considerations. Specifically, GL 95–05 requires the following: (1) enhanced eddy current inspection guidelines are implemented to provide consistency in voltage normalization; (2) 100 percent bobbin coil inspections are performed each cycle for all hot leg TSP intersections and all cold leg TSP intersections down to the lowest cold leg TSP with known ODSCC indications; and (3) rotating pancake coil (RPC) inspection of indications greater than 2 volts are performed to characterize the principal degradation as ODSCC. DCPP's proposed voltage-based repair criteria implementation practices meet the above requirements, and in some areas exceed them (for example, 100 percent bobbin coil inspections are routinely performed each cycle on every TSP intersection).

Implementation of voltage-based repair criteria at TSP intersections will decrease the number of tubes which must be repaired. Since the installation of tube plugs to remove ODSCC degraded tubes from service reduces RCS flow margin, voltage-based repair criteria implementation will help preserve the margin of RCS flow.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

NRC Project Director: William H. Bateman.

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

Date of amendment requests: February 27, 1997.

Description of amendment requests: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 by revising Technical Specifications (TS) 3/4.8.1.1, ''A.C. Sources—Operating,'' to clarify that emergency diesel generator (EDG) testing is initiated from standby conditions rather than ''ambient'' conditions. The associated TS Bases will be revised to discuss the temperature range that satisfies EDG standby conditions. This amendment also proposes to revise TS 3/4.3.2, ''Instrumentation—Engineering Safety Features Actuation System Instrumentation.'' This revision clarifies that when one or both of the first level load shed relays, or one or both of the second level undervoltage relays are inoperable, the associated EDG for that bus shall be declared inoperable.

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

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

The proposed changes to the technical specifications (TS) do not change the function or operation of any plant equipment or affect the response of that equipment if it is called upon to operate.

The proposed change to TS 4.8.1.1.2a.2 and the Bases will clarify the term ''ambient conditions'' as used in the emergency diesel generator (EDG) surveillance requirement. EDG testing will still be completed on a frequency commensurate with the current TS.

The proposed change to TS 3.3.2, Table 3.3–3, will permit time to restore the load shed first level undervoltage relays (FLURs) and second level undervoltage relays (SLURs) to operable status that is consistent with times allowed for outage of other safety-related equipment affecting one train of vital equipment. This proposed change maintains a high degree of equipment availability without requiring unnecessary initiation of a plant shutdown for partial equipment outages.

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

2. The proposed change does not create the possibility of a new or different kind of

accident from any accident previously evaluated.

The proposed change to TS 4.8.1.1.2a.2 and the Bases will clarify the term ''ambient conditions'' as used in the EDG surveillance requirement. EDG testing will still be completed on a frequency commensurate with the current TS, and will be more representative of the conditions under which the EDGs would be required to start in an accident condition.

The proposed change to TS 3.3.2, Table 3.3–3, will provide time to restore the load shed FLURs and SLURs to operable status that is consistent with times allowed for outage of other safety-related equipment affecting one train of vital equipment. The load shed FLUR and SLUR sets for one 4 kV bus only affect one train of vital equipment. If an accident occurred while the relays were inoperable, the redundant trains (two remaining EDGs and vital buses) would complete the safety function. The proposed allowed outage time (AOT) for the load shed FLURs and SLURs is bounded by the time allowed for an EDG supporting the vital 4 kV bus and is consistent with AOTs for other safety-related components.

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

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

The proposed change to TS 4.8.1.1.2a.2 and its Bases, clarifies the term ''ambient conditions'' as used in the EDG surveillance requirement. EDG testing will still be completed on a frequency commensurate with the current TS. Use of temperatures in the standby range result in no significant variation in EDG start times as indicated by the diesel vendor and by PG&E test results. Standby conditions are representative of actual starting conditions that would be in effect if the EDGs started in an accident.

The proposed change to TS 3.3.2, Table 3.3–3, will provide time to restore the load shed FLURs and SLURs to operable status that is consistent with times allowed for outage of other safety-related equipment affecting one train of vital equipment. If an accident occurred while the relays were inoperable, the redundant trains (two remaining EDGs and vital buses) would complete the safety function. The proposed change eliminates an unneccessary plant shutdown and associated risk due to shutdown transient. It prevents a transient that could require the EDGs at a time when less than all three EDGs would be operable.

Therefore, neither of the proposed changes involves a significant reduction in a margin of safety.

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: January 28, 1997.

Description of amendment request: The proposed amendment by Portland General Electric (PGE or the licensee) clarifies the administrative controls that are used for the revision and maintenance of the Certified Fuel Handler Training Program. The change allows the licensee to make changes to the certified fuel handlers program without prior NRC staff approval. The text of the proposed change is taken from the improved standard technical specifications, NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.''

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

In accordance with the requirements of 10 CFR 50.92, ''Issuance of amendment,'' this license amendment request is judged to involve no significant hazards consideration based upon the following:

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

The proposed change is a clarification of the method of control that will be used for the Certified Fuel Handler Training Program, and as such, is administrative in nature and has no impact on the probability or consequences of accidents previously evaluated. The physical structures, systems, and components of the facility and the operating procedures for their use are unaffected by this proposed clarification. The proposed administrative controls provide adequate confidence that personnel that perform the certified fuel handler functions will have been adequately trained for the changing conditions of the facility. Since the training program will prepare the operations personnel for fuel handling operations, including responses to abnormal events/ accidents, there will be no increase in the probability of occurrence or in the consequences of an accident previously evaluated.

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

This change ensures the qualifications of the operations personnel are commensurate with the tasks to be performed and the conditions to which they may be required to respond. This change does not affect plant equipment or the procedures for operating plant equipment and, therefore, 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.

This change ensures the qualification of the operations personnel are commensurate with the tasks to be performed and the conditions to which they may be required to respond. The assumptions for a fuel handling accident in the Fuel Building are not affected by the proposed change. The proposed amendment does not, therefore, involve a reduction in a margin of safety.

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

Local Public Document Room location: Branford Price Millar Library, Portland State University, 934 S.W. Harrison Street, P.O. Box 1151, Portland, Oregon 97207.

Attorney for the Licensees: Leonard A. Girard, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRR Project Director: Seymour H. Weiss.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: September 11, 1996.

Description of amendment request: The proposed amendment would permit operation with increased safety relief valve (SRV) and safety valve (SV) setpoint tolerance and permit operation up to 100% of rated power with a single inoperable SRV.

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

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

The proposed changes will permit operation with increased SRV and SV setpoint tolerance and permit operation up to 100% of rated power with a single inoperable SRV.

The valves are not related to the control rod system. The valves are not involved in the initiation of a Control Rod Drop Accident. The valves are part of the Reactor Vessel (RV) pressure boundary and their failure could initiate a LOCA [loss-of-coolant accident]. However, the proposed changes do not constitute a change in the design of the valves from a pressure boundary perspective. The proposed changes do not affect the probability of a LOCA initiated by valve failure. The valves are not a component, system, or structure involved in refueling operations. The valves and their as-found setpoint tolerance are not involved in the initiation of a Refueling Accident.

The design basis Main Steam Line Break is a complete severance of one main steam line outside the secondary containment. The SRVs and SVs are located inside primary containment and cannot cause a main steam line rupture outside secondary containment. The valves are not involved in the initiation of a design basis Main Steam Line Break. The probability or consequences of these accidents are not affected.

Attachment C [see application dated September 11, 1996] includes an analysis to demonstrate that margin exists to SV challenges during an Abnormal Operational Transient (AOT). For this purpose a Generator Load Rejection without Bypass (GLRWOBP) was identified as the limiting AOT. The results confirm that SV challenges would not occur with an inoperable SRV at rated power.

The current Technical Specification limit of 95% rated power or less with an inoperable SRV is therefore not required to prevent SV challenges during an AOT.

As discussed in Attachment C [see application], the impact of the proposed as-found SRV setpoint tolerance increase on SRV piping/supports and discharge loads to the Torus was evaluated. A mechanical loads analysis confirmed the integrity of these components, systems, and structures during SRV discharge with the proposed changes.

Attachment C [see application] provides an evaluation of the impact of the proposed changes on the consequences of the Loss of Coolant Accident and the Main Steam Line Break. The limiting LOCA event is a break in the recirculation loop, with a break area of 0.6 ft^2, at the pump discharge location, with a loss of one train of DC power as the single failure. For breaks in the recirculation line larger than 0.4 ft^2, the SRVs would not be challenged. Therefore, in assessing the impact of the proposed changes on 10CFR50.46 acceptance criteria, only recirculation line breaks less than 0.4 ft^2 were reevaluated. Results show that the 0.6 ft^2 recirculation line break remains the limiting LOCA event and it is not affected. The consequences of the limiting design basis LOCA are not increased by the proposed changes. The design basis accident for containment performance is a double-ended break in the recirculation pump suction. For this size break, the SRVs are not challenged. Therefore, the proposed changes do not have any effect on the design basis accident for containment performance. The design basis accident for radioactive material releases and radiological effects is a complete severance of one main steam line outside the secondary containment. For steam line breaks outside the containment, MSIVs [main steam isolation valves] close and terminate radiological releases outside the containment, SRVs are not challenged until after MSIV closure and isolation. Therefore, the proposed changes do not increase the radiological consequences of the design basis Main Steam Line Break.

The SRVs and SVs are designed to mitigate the consequences of malfunctions of equipment which result in a Nuclear System pressure increase. These abnormal operational transients are defined and analyzed in Section 14.5.1 of the VY [Vermont Yankee] FSAR [final safety analysis report]. The impact of the proposed changes on these abnormal operational transients was evaluated. Results are documented in Attachment C [see application] and show that applicable acceptance criteria are met provided operating MCPR [minimum critical power ratio] limits as specified in the COLR [core operating limit report] are adjusted to reflect the effects of the proposed changes. A hot channel analysis of the limiting delta CPR overpressure transient confirmed that a 0.02 increase in the operating MCPR limits bounds the combined effects of implementing the proposed changes in the current cycle. The operating MCPR limits in COLR have already been increased for the current cycle. Appropriate operating MCPR limits for future cycles will be determined from cycle-specific safety analyses performed with the approved changes.

Current practice regarding SRV setpoints is to assure plus or minus 1% tolerance is met as required by the ASME [American Society of Mechanical Engineers] Boiler & Pressure Vessel Code referenced in Technical Specification Surveillance Requirement 4.6.E.2. As-left setpoints always meet the plus or minus 1% tolerance. The safety analysis in Attachment C [see application] demonstrates that as-found setpoints within plus or minus 3% are acceptable. However, valves re-installed after testing will continue, as previously, to meet plus or minus 1% tolerance as required by the ASME Boiler & Pressure Vessel Code. Thus, the probability of SRV actuation (and the associated risk of failure to reseat properly) is not increased by the proposed change.

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

The proposed changes will permit operation with increased Safety Relief Valve (SRV) and Safety Valve (SV) setpoint tolerance and permit operation up to 100% of rated power with a single inoperable SRV. The proposed changes:

(1) do not constitute a change in the design of the valves;

(2) will not cause the valve or associated systems and structures to be operated beyond their original design envelopes; and,

(3) do not involve new plant equipment.

Therefore, this amendment does not create the possibility of a new or different kind of accident.

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

Technical Specification Basis 3.6 and 4.6D identifies the minimum critical power ratio (MCPR) safety limit. Operational restraints on MCPR are placed in the COLR to assure no violation of the MCPR safety limit during AOTs. The impact of the proposed changes on MCPR limits was determined by performing a hot channel analysis for the overpressure transient which yields the largest transient drop in CPR [critical power ratio] (delta CPR). Results are documented in Attachment C [see application], and show that a 0.02 increase in the operating MCPR limits bounds the combined effects of the proposed changes and assures the MCPR safety limit is not violated during AOTs. The margin of safety defined by the MCPR safety limit is not reduced.

Technical Specification Basis 3.6 and 4.6D also identifies the ASME Boiler and Pressure Vessel Code Section III-A limit which permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig). This margin of safety is not reduced by the proposed changes. Attachment C [see application] documents new overpressure transient analysis with results that demonstrate the ASME overpressure limit of 110% of design is met. This license amendment request does not propose to reduce the margin of safety defined by the ASME Boiler & Pressure Vessel Code limit.

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

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: R. K. Gad, III, Ropes and Gray, One International Place, Boston, MA 02110–2624.

NRC Project Director: Patrick D. Milano, Acting.

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

Date of amendment request: February 3, 1997 as supplemented March 18, 1997.

Description of amendment request: The proposed change to Technical Specification 4.15.B.1 is administrative in nature in that it revises the Technical Specifications (TS) to be consistent with the NRC-approved inservice inspection program. In addition, three TS pages which were previously approved by NRC, and which were inadvertently omitted in an earlier amendment (amendments 40 and 39 for units 1 and 2, respectively), are being reissued.

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 Surry Units 1 and 2 in accordance with the proposed Technical Specifications change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change is administrative in nature, and station operations are not being affected. The ASME Section XI Code requirements are thoroughly established, reviewed and approved by ASME, the industry and ultimately endorsed by the NRC for inclusion into 10 CFR 50.55a. Updates to the Code reflect advances in technology and consider information obtained from plant operating experience to provide enhanced inspection and examination techniques for pipe welds. Therefore, performing weld examinations for the pipe in our augmented inspection program to the requirements of the 1989 edition of the ASME Section XI Code provides a regulatory acceptable and adequate level of assurance that the integrity of the pipe will be maintained. By not referencing a specific Code edition in the Technical Specifications, our examinations for pipe in the augmented inspection program will consistently be performed to the Code of record, consistent with the requirements [of] 10 CFR 50.55a. Consequently, the probability or consequences of an accident previously evaluated are not increased.

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

As noted above, the proposed change is administrative in nature, and no new accident precursors are being introduced. Since the augmented inspection program will continue to be performed to NRC approved ASME Section XI Code requirements, adequate assurance is provided to ensure the integrity of the pipe. Consequently, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Performing weld examinations to the Code of record is prudent, consistent with accepted industry and regulatory requirements, and provides adequate assurance that piping integrity will be maintained. The use of a general ASME Section XI Code reference in Technical Specification 4.15.B.1 is consistent with the existing wording in Technical Specifications 4.15.A and C, and ensures that weld examinations are being consistently performed to the currently approved edition of the ASME Section XI Code. This is an administrative change and as such 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. The staff notes that the reissuance of three TS pages is a purely administrative matter which involves no significant hazards consideration and which has been considered previously. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

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

NRC Project Director: Mark Reinhart, Acting.

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

Dates of amendment requests: June 4, 1996, as supplemented August 5, September 26, October 21, November 13, November 20, and December 2, 1996, and January 16, March 5, and March 20, 1997 (TSCR 188 and 189).

Description of amendment requests: The proposed amendments would revise License Nos. DPR–24 and DPR–27 to add commitments for control room habitability and revise Technical Specification (TS) Sections 15.1, ''Definitions,'' 15.2.1, ''Safety Limit, Reactor Core,'' 15.2.3, ''Limiting Safety System Settings and Protective Instrumentation,'' Section 15.3.1, ''Reactor Coolant System,'' 15.3.4, ''Steam and Power Conversion System,'' 15.3.5, ''Instrumentation System, 15.4.1, ''Operational Safety Review,'' 15.5.3, ''Design Features—Reactor,'' and 15.6.9, ''Plant Reporting Requirements,'' and modify the bases for Section 15.2.2, ''Safety Limit, Reactor Coolant System Pressure,'' and Section 15.3.1.C, ''Maximum Coolant Activity,'' to incorporate changes associated with the operation of Point Beach Nuclear Plant (PBNP), Unit 2, with replacement steam generators. The new analyses performed for replacing Unit 2 steam generators resulted in changes to the reactor core safety limits and protective instrumentation setpoints for Unit 1 as well as Unit 2. Calculations are based on operation at either 2000 psia or 2250 psia and an average temperature limit of greater than or equal to 557 degrees Fahrenheit and less than or equal to 573.9 degrees Fahrenheit. New dose calculations were performed based on new setpoints for low-low steam generator water level, new values of primary and secondary steam generator volumes, and revised accident analyses for steam generator tube rupture, main steam line break, locked rotor, and control rod ejection. Additional license conditions are proposed to document the commitments made to improve habitability of the control room so that dose limits do not exceed 10 CFR Part 50, Appendix A, General Design Criterion 19, without relying on the use of potassium iodide pills and/or self-contained breathing apparatus. The original applications were previously noticed in the **Federal Register** on July 3, 1996 (61 FR 34903 and 34904).

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.

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

The proposed TS changes reflect the replacement of steam generators at PBNP, including new analyses and setpoints, and a different standard and acceptance criteria for Dose Equivalent I–131. The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. The staff independently performed an evaluation of the dose consequences for steam generator tube rupture, main steam line break, locked rotor accident, and a rod ejection accident. The staff determined there are no significant increases in dose for the low population zone or the exclusion area boundary. The licensee had not previously analyzed these accidents for control room habitability. As a result of the proposed changes, limiting control room doses will require compensatory measures, use of potassium iodide and self-contained breathing apparatus, which have been previously approved, until such time that the control room ventilation design is improved. The commitments to improve control design/operation are included as license conditions. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Installation of new steam generators, with a small increase in primary side volume and new setpoints for instrumentation, does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with the replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. These changes do not affect any of the parameters or conditions that contribute to initiation of any accidents. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed setpoints maintain the margin to safe operation of Unit 2 with the replacement steam generators. In order to maintain one set of safety analyses for both units, the analyses for operation of Unit 2 with replacement steam generators were performed to encompass the operation of Unit 1. Therefore, the proposed changes apply to the operation of both units and maintain the margin of safety for each. Compensatory measures will ensure control room doses remain within the dose guidelines in 10 CFR Part 50, Appendix A, General Design Criterion 19, until such time as the control ventilation system design/operation is revised. Therefore, the proposed changes do not involve 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 requests involve 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: John N. Hannon.

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

Date of amendment request: September 30, 1996, as supplemented November 26, and December 12, 1996, February 13, and March 5, 1997 (TSCR 192).

Description of amendment request: The proposed amendments would revise License Nos. DPR–24 and DPR–27 to add commitments for control room habitability and revise Technical Specification (TS) Sections 15.3.3, ‘‘Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray,’’ TS 15.3.7, ‘‘Auxiliary Electrical Systems,’’ 15.5.2, ‘‘Design Features-Containment,’’ and associated TS Bases to reflect proposed changes to the limiting conditions for operation, action statements, allowable outage times, and design specifications for the Point Beach Nuclear Plant (PBNP) TS associated with the containment accident fan coolers, service water equipment (pumps and piping), component cooling water pumps, and normal and emergency power supplies. Specifically, these proposed changes increase the number of service water pumps and component cooling water pumps required to be operable, change the description of the service water system to define three separate loops, modify the limiting conditions for operation of the containment cooling and iodine removal systems and the component cooling water and service water systems, modify the auxiliary electrical system requirements, modify the associated TS Bases, and change the design value for each containment ventilation/air coolers from 55,600 Btu/sec to 41,700 Btu/sec. The original application was previously noticed in the **Federal Register** on November 19, 1996 (61 FR 58905).

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.

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

The proposed changes involve components currently installed in the facilities and reflect current capabilities of this equipment. Increasing the number of service water and component cooling water pumps required to be operable, changing the service water header definitions and modifying the limiting conditions for operation for service water and component cooling water, and modifying the requirements for the 4160/480-volt safeguards buses does not increase the probabilities of any accidents currently evaluated in the final safety analysis report (FSAR). The probabilities of accidents previously evaluated in the FSAR are based on the probability of initiating events for these accidents. Initiating events for accidents previously evaluated for Point Beach include: Control rod withdrawal and drop, CVCS [chemical volume and control system] malfunction (boron dilution), startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, losses of reactor coolant flow, loss of external electrical load, loss of normal feedwater, loss of all AC power to the auxiliaries, turbine overspeed, fuel handling accidents, accidental releases of waste liquid or gas, steam generator tube rupture, steam pipe rupture, control rod ejection, and primary coolant system ruptures. The change to the heat removal capability of the containment ventilation/air coolers from 55,600 Btu/sec to 41,700 Btu/sec was evaluated to ensure that containment design is not challenged. Therefore, the proposed changes do not affect the probability of occurrence or the consequences of any accident previously evaluated in the FSAR. During review of the proposed changes, the staff determined that other changes made to the operation of the containment spray system and the control room ventilation design and operation could affect the doses associated with a loss-of-coolant accident. The staff has determined that there is no significant increase in offsite doses. As a result of the proposed changes and current plant design, limiting control room doses will require compensatory measures, use of potassium iodide and self-contained breathing apparatus, which have been previously approved, until such time that the control room ventilation design/operation is improved. The commitments to improve control design/operation are included as license conditions.

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

The proposed changes do not introduce any new accidents from any previously evaluated. Failures for the systems affected by the proposed changes, service water system, component cooling water system, containment ventilation/air cooling units, and the 4160/480-volt safeguards buses are factored into the accident analyses included in the FSAR. No new or different kinds of accidents are created since no new or different accident initiators or sequences are involved. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the Point Beach FSAR.

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

The proposed changes provide the appropriate limiting conditions for operation, action statements, allowable outage times, and design specifications for service water, component cooling water, containment cooling, and normal and emergency power supplies. This ensures that the safety systems that protect the reactor and containment will operate as required. The impact of changes to design and operation of affected systems do not affect the reactor and containment design. Therefore, the margins of safety for Point Beach are not being reduced because the design and operation of the reactor and containment are not being changed and the safety systems and limiting conditions of operation for these safety systems that provide their protection that are being changed will continue to meet the requirements for accident mitigation for PBNP. Compensatory measures will ensure control room doses remain within the dose guidelines in 10 CFR Part 50, Appendix A, General Design Criterion 19, until such time as the control ventilation system design/

operation is revised. Therefore, the proposed changes will not create 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 requests involve 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: John N. Hannon.

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.

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

Date of amendment request: March 7, 1997.

Description of amendment request: The proposed amendments would revise Technical Specification 3/4.7.1.6 and Section 15.6.3 of the Updated Final Safety Analysis Report to require four instead of three steam generator pressure operated relief valves operable.

Date of publication of individual notice in **Federal Register:** March 13, 1997 (62 FR 11931).

Expiration date of individual notice: April 14, 1997.

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

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of amendment request: March 17, 1997.

Brief description of amendment request: The proposed amendment would modify the Design Features Section 5.3.1 of the Technical Specifications to reflect the Atrium-10 design and would include a Siemens Power Corporation topical report reference in Section 6.9.3.2 to reflect mechanical design criteria for this fuel. This change would allow this fuel to be loaded and maintained in the core only under Condition 5, (refueling).

Date of publication of individual notice in **Federal Register:** March 25, 1996 (62 FR 14167).

Expiration date of individual notice: April 24, 1997.

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

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.

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

Date of application for amendments: May 2, 1995, as supplemented by letter dated March 7, 1996.

Brief description of amendments: These amendments modify the licenses to authorize incorporation in the Updated Final Safety Analysis Report (UFSAR) of certain changes to the description of the facilities involving a revised large-break loss of coolant accident (LOCA) analysis that addresses a previously unanalyzed release path through the steam generators to the atmosphere.

Date of issuance: March 17, 1997.

Effective date: March 17, 1997, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 1—111; Unit 2—103; Unit 3—83.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Operating Licenses and Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register:** December 6, 1995 (60 FR 62487). The March 7, 1996, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 17, 1997. No significant hazards consideration comments received: No.

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

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: January 29, 1997, as supplemented February 6, and February 21, 1997.

Brief description of amendment: The amendment adds a new Technical

Specification 3.0.5 to provide guidance for returning equipment to service under administrative controls for the sole purpose of performing testing to demonstrate operability.

Date of issuance: March 17, 1997.

Effective date: March 17, 1997.

Amendment No.: 69.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** February 12, 1997 (62FR6569).

The February 6, and February 21, 1997 letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 1997.

No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: January 10, 1997, as supplemented January 31, February 20, and March 3, 1997.

Brief description of amendment: The amendment revises Technical Specification 4.8.1.1.2 to clarify pressure testing requirements for the isolable and non-isolable portions of the diesel fuel oil piping.

Date of issuance: March 19, 1997.

Effective date: March 19, 1997.

Amendment No.: 70.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 5, 1997 (62 FR 5490). The January 31, February 20, and March 3, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 19, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of application for amendments: November 4, 1996 and supplemented February 5, 1997.

Brief description of amendments: The amendments revise Technical Specification Section 4.7.13.1.c to eliminate the requirement that the 18-month Standby Shutdown System diesel generator inspection be performed only during shutdown of both reactors.

Date of issuance: March 13, 1997.

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

Amendment Nos.: Unit 1—157—Unit 2—149.

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

Date of initial notice in **Federal Register**: December 4, 1996 (61 FR 64383) The supplemental letter dated February 5, 1997, provided additional information that did not change the scope of the November 4, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 13, 1997.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: January 13, 1997.

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

Date of issuance: March 21, 1997.

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

Amendment Nos.: Unit 1—173—Unit 2—155.

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

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6575) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, North Carolina 28223–0001.

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: February 15, 1996, as supplemented by letter dated February 18, 1997.

Brief description of amendments: The amendments add operability and surveillance requirements regarding operation and testing of the Keowee Hydro Station to the Oconee Technical Specifications.

Date of Issuance: March 20, 1997.

Effective date: As of the date of issuance to be implemented within 30 days. Implementation shall include revision of the Selected Licensee Commitment manual to incorporate the Keowee Hydro units' commercial power operating restrictions curves in accordance with the application for the amendments.

Amendment Nos.: Unit 1—222; Unit 2—222; Unit 3—219.

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

Date of initial notice in **Federal Register**: March 27, 1996 (61 FR 13523) The February 18, 1997, letter provided clarifying information that did not change the scope of the February 15, 1996, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691.

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

Date of application for amendments: September 18, 1992, as supplemented October 6, 8, 15, 23, and November 13 and 20, 1992, March 5, May 24, June 10, and December 20, 1993, April 6 and July 28, 1995, and September 11, October 1, December 13, 19 and 23, 1996.

Brief description of amendments: The amendments modify the Facility Operating Licenses, Technical Specifications, Environmental Protection Plan, and Antitrust conditions to add Southern Nuclear Operating Company, Inc., as operator of the facilities, with exclusive responsibility and control over its physical construction, operation, and maintenance. The antitrust license conditions divorce Southern Nuclear from marketing or brokering power or energy from the Hatch Plant and holds Georgia Power Company accountable for the actions of its agent, Southern Nuclear, to the extent Southern Nuclear's actions contravene the Hatch antitrust license conditions. An Order Approving Southern Nuclear Operating Company, Incorporated, As Exclusive Operator was included along with the issuance of the amendments.

Date of issuance: March 17, 1997.

Effective date: To be implemented within 60 days of the date of issuance.

Amendment Nos.: 203 and 144.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications and Operating Licenses.

Date of initial notice in **Federal Register**: October 14, 1992 (57 FR 47131). The October 6, 8, 15, 23, and November 13 and 20, 1992, March 5, May 24, June 10, and December 20, 1993, April 6 and July 28, 1995, and September 11, October 1, December 13, 19 and 23, 1996, letters, did not change the scope of the September 18, 1992, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 17, 1997, and an Environmental Assessment dated October 27, 1992.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

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

Date of application for amendments: October 7, 1996.

Brief description of amendments: The amendments revise Surveillance Requirements (SRs) 3.1.7.7 and 3.4.3.1, and Limiting Conditions for Operation 3.4.3, 3.5.1, and 3.6.1.6 to increase the nominal mechanical pressure relief setpoints for all of the 11 safety/relief valves (SRVs) to 1150 psig and allow operation with one SRV and its associated functions inoperable. The change will reduce the potential for SRV pilot leakage and the potential for forced outages due to an inoperable SRV during a fuel cycle.

Date of issuance: March 21, 1997.

Effective date: As of the date of issuance to be implemented for Unit 1 prior to startup from its refueling outage scheduled for fall 1997; and for Unit 2 prior to startup from its refueling outage currently scheduled for March 15, 1997.

Amendment Nos.: 204 and 145.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 129). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

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

Date of application for amendments: October 29, 1996, as supplemented February 19, 1997.

Brief description of amendments: The amendments revise the Technical Specifications associated with the installation of a digital Power Range Neutron Monitoring system.

Date of issuance: March 21, 1997.

Effective date: As of the date of issuance to be implemented for Unit 1 prior to its startup from the fall of 1997 refueling outage; and implemented for Unit 2 prior to its startup from the spring of 1997 refueling outage.

Amendment Nos.: 205 and 146.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 130). The February 19, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

GPU Nuclear Corporation, et al., Docket No. 50–219, Oyster Creek Nuclear

Date of application for amendment: November 27, 1996 (TSCR 232).

Brief description of amendment: The amendment changes the acceptance criteria for the individual cell voltage from 2.0v to 2.09v, the frequency for battery specific gravities to implement the recommendations of IEEE 450–1995, deletes surveillance 4.7.B.4.d, and adds a clarifing phrase ''while on a float charge . . .'' where appropriate.

Date of Issuance: March 24, 1997.

Effective date: March 24, 1997.

Amendment No.: 189.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: February 12, 1997 (62 FR 6576) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 24, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendment request: November 20, 1996, as supplemented by letters dated February 20, 1997, and March 25, 1997.

Brief description of amendment: The amendment revises Section 5.2 of the Fort Calhoun Station technical specifications to relocate controls for working hours to the Updated Safety Analysis Report.

Date of issuance: March 27, 1997.

Effective date: March 27, 1997.

Amendment No.: 181.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications and operating license.

Date of initial notice in **Federal Register**: January 2, 1997 (62 FR 131) The February 20, 1997, and March 25, 1997, supplemental letters provided additional clarifying information that did not change the portion of the initial no significant hazards consideration determination that addressed this proposed change.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 27, 1997.

No significant hazards consideration comments received: No.

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

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997

Brief description of amendment: The amendments revise Section 6.0 (Administrative Controls) of the Hope Creek TS to: (1) Relocate the requirements of Section 6.5 (Station Operations Review Committee, Nuclear Safety Review and Audit, and Technical Review and Control) to the Quality Assurance Program, (2) replace specific management titles with generic management functional positions, (3) change Operating Engineer to Assistant Operations Manager, (4) require a Senior Reactor Operator license be held by either the Operations Manager or one of the Assistant Operations Managers, and (5) correct some typographical errors in Section 6.0.

Date of issuance: March 21, 1997.

Effective date: As of date of issuance and shall be implemented within 60 days.

Amendment No.: 97.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications and the license.

Date of initial notice in **Federal Register**: February 14, 1996 (61 FR 5817).

The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the original notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: October 25, 1996, as supplemented by letters dated December 4, 1996, and January 24, 1997.

Brief description of amendment: This amendment changes Hope Creek Technical Specification (TS) 3/4.1.3.5, ''Control Rod Scram Accumulator,'' in order to: 1) Permit a separate entry into a TS action statement for each inoperable control rod; 2) provide more specific applicability for required actions in Operational Condition 1 or 2 with one inoperable control rod scram accumulator (reactor pressure of ≥ 900 psig would be specified); 3) provide more specific actions (verify charging water pressure) for two or more inoperable control rod scram accumulators when reactor pressure is ≥ 900 psig; 4) provide more specific actions when reactor pressure is < 900 psig and one or more control rod scram accumulators are inoperable (verify insertion of control rods associated with inoperable accumulators and verify that charging water header pressure is ≥ 940 psig); 5) provide specific actions in Operational Condition 5 with one or more withdrawn control rods inoperable; and 6) eliminate the requirements to perform an 18-month channel functional test of the leak detectors and the 18-month channel calibration of the pressure detectors.

Date of issuance: March 26, 1997.

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

Amendment No.: 98.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 4, 1996 (61 FR 64394) The December 4, 1996, and January 24, 1997, supplements did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

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

Date of application for amendments: January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997.

Brief description of amendments: The amendments revise Section 6.0 (Administrative Controls) of the Salem TS to: 1) relocate the requirements of Section 6.5 (Station Operations Review Committee, Nuclear Safety Review and Audit, and Technical Review and Control) to the Quality Assurance Program, 2) replace specific management titles with generic management functional positions, 3) change Operating Engineer to Assistant Operations Manager, 4) require a Senior Reactor Operator license be held by either the Operations Manager or one of the Assistant Operations Managers, and 5) correct some typographical errors in Section 6.0.

Date of issuance: March 21, 1997.

Effective date: Both units, as of date of issuance, to be implemented within 60 days.

Amendment Nos.: 192 and 175.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications and the license.

Date of initial notice in **Federal Register**: February 14, 1996 (61 FR 5818) The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor the original notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 21, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama

Date of amendment request: December 26, 1997, as supplemented by letter dated February 6, March 7, and March 21, 1997.

Brief Description of amendment: The amendment changes Technical Specification 3/4.4.6, ''Steam Generators'' and associated Bases to implement the voltage-based alternate repair criteria for steam generator tubes in Farley Unit 1 in accordance with

Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

Date of issuance: March 24, 1997.

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

Amendment No.: 124.

Facility Operating License Nos. NPF–2 and NPF–8: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 29, 1997 (62 FR 4353) By letter dated February 6, 1997, the licensee submitted additional information to clarify the changes to the proposed repair criteria, which did not change the scope of the December 26, 1996, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 24, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: January 10, 1997, as supplemented by letter dated February 24, 1997.

Brief Description of amendments: The amendments revise the Technical Specifications (TS) to incorporate the latest revised topical reports governing the installation of laser welded steam generator tube sleeves. In addition, the reference to a one-cycle implementation of L*, which expired at the last Unit 2 outage was deleted from the Unit 2 TS.

Date of issuance: March 24, 1997.

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

Amendment Nos.: 125 and 119.

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

Date of initial notice in **Federal Register**: January 29, 1997 (62 FR 4355) The February 24, 1997, letter provided clarifying information that did not change the original application and the initial proposed no significant hazards consideration determination published in the **Federal Register** on January 29, 1997 (62 FR 4355).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 24, 1997.

No significant hazards consideration comments received: No.

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

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

Date of amendments request: September 30, 1996.

Brief description of amendments: The amendments revise Technical Specifications (TS) 3/4.1.1.1, 3/4.1.1.2, 3/4.1.1.3, 3/4.1.3.5, 3.1.3.6, 3.2.1, 3.2.2 and 3.2.3 and associated Bases to remove certain cycle-specific parameter limits from the TS and relocate them to the Core Operating Limits Report.

Date of issuance: March 25, 1997.

Effective date: As of the date of issuance to be implemented for Unit 1 prior to entry into Mode 5 following the next scheduled refueling outage, which should begin in March 1997; for Unit 2 prior to entry into Mode 5 following the refueling outage scheduled to begin in March 1998.

Amendment Nos.: 126 and 120.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications and License Conditions.

Date of initial notice in **Federal Register**: November 6, 1996 (61 FR 57491) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 25, 1997.

No significant hazards consideration comments received: No.

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

Southern California Edison Company, et al., Docket No. 50–362, San Onofre Nuclear Generating Station, Unit No. 3, San Diego County, California

Date of application for amendment: February 18, 1997, as supplemented by letter dated February 21, 1997.

Brief description of amendment: The amendment defers implementation of Surveillance Requirement 3.3.5.6 of Technical Specifcation 3.3.5, ''Engineered Safety Features Actuation System (ESFAS) Instrumentation,'' until the next SONGS Unit 3 shutdown, which will be no later than the upcoming Cycle 9 refueling outage (currently scheduled for April 12, 1997).

Date of issuance: March 17, 1997.

Effective date: March 17, 1997.

Amendment No.: 127

Facility Operating License No. NPF–15: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: Yes (62 FR 9001 dated February 27, 1997). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 31, 1997, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The February 21, 1997, letter provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 17, 1997.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

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

Date of application for amendment: February 14, 1997.

Brief description of amendment: This amendment revises Technical Specification (TS) Section 3/4.5.2, ''Emergency Core Cooling Systems, ECCS Subsystems—$T_{avg} \geq 280°F$.'' Surveillance requirement 4.5.2.f would be modified to state that opening and closing of the inspection port on the watertight enclosure for the decay heat valve pit would not require this surveillance procedure to be performed. This amendment also revises the applicable TS bases.

Date of issuance: March 24, 1997.

Effective date: Immediately, and shall be implemented no later than 120 days after issuance.

Amendment No.: 215.

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (62 FR 8783 dated February 26, 1997). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by March 30, 1997, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated March 24, 1997.

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

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, Ohio 43606.

United States Department of Commerce, National Institute of Standards and Technology, Docket No. 50–184, NIST Test Reactor

Date of application for amendment: January 17, 1997.

Brief description of amendment: This amendment revises the Technical Specifications to change the name of the Reactor Radiation Division to the NIST Center for Neutron Research and the Chief, Radiation Division to Director, NIST Center for Neutron Research.

Date of issuance: March 31, 1997.

Effective date: March 31, 1997.

Amendment No.: 8.

Amended Facility License No. TR–5: This amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 26, 1997 (62 FR 8801). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 31, 1997.

No significant hazards consideration comments received: No.

Local Public Document Room Location: N/A.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

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

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By May 9, 1997, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director)*: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, and to the attorney for the licensee.

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

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

Date of application for amendments: December 27, 1996, as supplemented by letter dated March 18, 1997.

Brief description of amendments: The amendments modify the licenses to authorize incorporation of certain changes to the description of the facilities involving offsite power sources into the Updated Final Safety Analysis Report (UFSAR) for the Palo Verde Nuclear Generating Station (PVNGS).

Date of issuance: March 26, 1997.

Effective date: March 26, 1997, to be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1—112; Unit 2—104; Unit 3—84.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the operating licenses and the Updated Final Safety Analysis Report.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 26, 1997.

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

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

NRC Project Director: William H. Bateman.

Commonwealth Edison Company, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: March 26, 1997, as supplemented on March 27, 1997.

Brief description of amendments: The proposed amendments provided (1) An evaluation of the Unreviewed Safety Question (USQ) involving the control room operator dose resulting from error in the secondary containment volume, (2) a change in Technical Specification (TS) 4.7.P.2.b and 4.7.P.3 values for the allowed methyl iodide penetration for the standby gas treatment charcoal adsorbers, and (3) change of TS 5.2.C to reflect the new calculated free volume of the secondary containment.

Date of Issuance: March 27, 1997.

Effective date: March 27, 1997.

Amendment Nos.: 175, 171.

Facility Operating License Nos. DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendments, finding of emergency circumstances and final determination of no significant hazards consideration are contained in a Safety Evaluation dated March 27, 1997.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

NRC Project Director: Robert A. Capra.

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

Date of application for amendments: January 29, 1997, as supplemented February 11, 12, March 7, 10, 11, 19, and 20, 1997.

Brief description of amendments: The amendments authorize Northern States Power Company to continue operation of Prairie Island Units 1 and 2 on an interim basis, through the incorporation of three license conditions into its licenses, until a seismically qualified emergency cooling water source is provided that will provide the basis to extend the time for operator post-seismic cooling water load management. This could be done either through a seismic evaluation of the intake canal, physical modifications to the intake canal or plant, or some combination of the two.

Date of issuance: March 25, 1997.

Effective date: March 25, 1997, with implementation of License Condition 1 prior to Unit 2 entering Mode 2, with implementation of the requirements of License Condition 2 by July 1, 1997, and December 31, 1998, and with implementation of License Condition 3 at the next updated safety analysis report update following completion of License Condition 2, but no later than June 1, 1999.

Amendment Nos.: 128 and 120.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the licenses.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (62 FR 5857 dated February 7, 1997). This notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. The notice also provided for an opportunity to request a hearing by March 10, 1997, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendments. Because of the significant revisions to the licensee's original application, NRC also published a public notice of the proposed amendments, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed finding be provided to the staff by close of business on March 20, 1997. The notice was published in the St. Paul Pioneer Press on March 15, 1997, the Minneapolis Star Tribune on March 16, 1997, and the Red Wing Republican Eagle on March 17, 1997. No comments have been received. The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated March 25, 1997.

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

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

NRC Project Director: John N. Hannon.

Dated at Rockville, Maryland, this 2nd day of April, 1997.

For the Nuclear Regulatory Commission.

Jack W. Roe,

Director, Division of Reactor Projects III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 97–8916 Filed 4–8–97; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–219, License No. DPR–16]

Oyster Creek Nuclear Generating Station; Issuance of Final Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), has granted in part and denied in part Petitions, dated September 19, 1994, and supplemented by a letter dated December 13, 1994, submitted by Messrs. Paul Gunter and William deCamp, Jr. (Petitioners) on behalf of Oyster Creek Nuclear Watch, Reactor Watchdog Project, and Nuclear Information and Resource Service. Petitioners requested that the NRC take immediate action with regard to Oyster Creek Nuclear Generating Station (OCNGS) operated by GPU Nuclear Corporation (GPU or licensee). By letter dated December 13, 1994, Petitioners supplemented the Petition dated September 19, 1994.

Specifically, the Petition of September 19, 1994, requested that the NRC (1) immediately suspend the OCNGS operating license until the licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the licensee has analyzed and mitigated any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements, and to promptly take appropriate mitigative action if the unit is not in compliance.

The supplemental Petition, in addition to providing more information on the original request, requested that the NRC (1) suspend the OCNGS operating license until the Petitioners' concerns regarding cracking are addressed, including inspection of all reactor vessel internal components and other safety-related systems susceptible to intergranular stress-corrosion cracking and completion of any and all necessary repairs and modification; (2) explain the discrepancies between the response of the NRC staff dated October 27, 1994, to the Petition of September 19, 1994, and time-to-boil calculations for the FitzPatrick plant; (3) require GPU to produce documents for evaluation of the time-to-boil calculation for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class IE systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the Petition; and (6) treat the Petitioners' letter of December 13, 1994, as a formal appeal of the denial of their request of September 19, 1994, to immediately suspend the OCNGS operating license.

The Director of the Office of Nuclear Reactor Regulation has granted requests (3), with the exception of suspending OCNGS operating license which was previously denied, and in part (4) of the Petition of September 19, 1994, and requests (2), (3), and (4) of the supplemental Petition of December 13, 1994. The reasons for these decisions are explained in the ''Final Director's Decision Under 10 CFR 2.206: (DD–97–08), the complete text of which follows this notice. The decision and the documents cited in the decision are available for public inspection and copying at the Commission's Public