The proposed action would also exempt the licensee from the requirements to maintain emergency procedures for each area in which this licensed SNM is handled, used, or stored to ensure that all personnel withdraw to an area of safety upon the sounding of the alarm, to familiarize personnel with the evacuation plan, and to designate responsible individuals for determining the cause of the alarm, and to place radiation survey instruments in accessible locations for use in such an emergency.

The proposed action is in accordance with the licensee's application for exemption dated December 5, 1997.

The Need for the Proposed Action

The purpose of 10 CFR 70.24 is to ensure that if a criticality were to occur during the handling of SNM, personnel would be alerted to that fact and would take appropriate action. At a commercial nuclear power plant, the inadvertent criticality with which 10 CFR 70.24 is concerned could occur during fuel handling operations. The SNM that could be assembled into a critical mass at a commercial nuclear power plant is in the form of nuclear fuel; the quantity of other forms of SNM that is stored on site is small enough to preclude achieving a critical mass. Because the fuel is not enriched beyond 5.0 weight percent Uranium-235 and because commercial nuclear plant licensees have procedures and features designed to prevent inadvertent criticality, the staff has determined that it is unlikely that an inadvertent criticality could occur due to the handling of SNM at a commercial power reactor. The requirements of 10 CFR 70.24, therefore, are not necessary to ensure the safety of personnel during the handling of SNM at commercial power reactors.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed action and concludes that there is no significant environmental impact if the exemption is granted. Inadvertent or accidental criticality will be precluded through compliance with the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specifications (TS), the design of the fuel storage racks providing geometric spacing of fuel assemblies in their storage locations, and administrative controls imposed on fuel handling procedures. TS requirements specify reactivity limits for the fuel storage racks and minimum spacing between the fuel assemblies in the storage racks.

Appendix A of 10 CFR Part 50, ''General Design Criteria for Nuclear Power Plants,'' Criterion 62, requires that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically-safe configurations. This is met at Sequoyah Nuclear Plant, Units 1 and 2, as identified in the TS and the Updated Final Safety Analysis Report (UFSAR). Sequoyah TS Section 5.6.1.2 states that the new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U–235 enrichment less than or equal to 5.0 weight percent, while maintaining a k-effective of less than or equal to 0.98 under the most reactive condition. UFSAR Section 9.1.1, New Fuel Storage, for both Units 1 and 2 specify that the fuel racks are designed to provide sufficient spacing between fuel assemblies to maintain a subcritical (k-effective less than or equal to 0.98) array assuming the most reactive condition, and under all design loadings including the safe shutdown earthquake. The UFSAR also specifies that the new fuel racks are designed to preclude the insertion of a new fuel assembly between cavities.

The proposed exemption would not result in any significant radiological impacts. The proposed exemption would not affect radiological plant effluent nor cause any significant occupational exposures since the TS design controls (including geometric spacing of fuel assembly storage spaces) and administrative controls preclude inadvertent criticality. The amount of radioactive waste would not be changed by the proposed exemption.

The proposed exemption does not result in any significant nonradiological environmental impacts. The proposed exemption involves features located entirely within the restricted area as defined in 10 CFR Part 20. It does not affect non-radiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant non-radiological environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission has concluded that there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed exemption, the staff considered denial of the requested exemption. Denial of the request would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the ''Final Environmental Statement Related to the Sequoyah Nuclear Plant, Unit Nos. 1 and 2,'' dated February 13, 1974.

Agencies and Persons Consulted

In accordance with its stated policy, on January 30, 1998, the Commission staff consulted with the State of Tennessee Official (Joelle Key) regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated December 5, 1997, which is available for public inspection at the Commission's Public Document Room, which is located at The Gelman Building, 2120 L Street, NW., Washington, D.C., and at the local public document room located at the Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee.

Dated at Rockville, Maryland, this 17th day of March 1997.

For the Nuclear Regulatory Commission.

Frederick J. Hebdon,

Director, Project Directorate II–3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 98–7812 Filed 3–24–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 2, 1998, through March 13, 1998. The last biweekly notice was published on March 11, 1998 (63 FR 11913).

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 and Directives Branch, Division of Administration 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 April 24, 1998, 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. 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 No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: February 23, 1998.

Description of amendment request: The amendment request proposes changes to the Brunswick Steam Electric Plant Unit 1 Technical Specifications (TS) in support of Cycle 12 operation, including a change to the Minimum Critical Power Ratio safety limit (safety limit MCPR) to a value equivalent to the generic safety limit MCPR for General Electric type GE–13 fuel. The request would additionally remove a footnote limiting the stated value for the safety limit MCPR to a specific fuel cycle and reference to an NRC safety evaluation documenting acceptance of methods used for determining the current cycle safety limit MCPR. The amendment request is provided both in the format of the current TS as well as improved Standard Technical Specifications (iSTS). The Brunswick licensee applied for conversion to ISTS on November 1, 1996, as supplemented on October 13, 1997, and February 26, 1998, and that application is currently undergoing NRC staff review. For iSTS, the licensee has proposed two safety limits MCPR, one pertaining to two-recirculation loop operation and the other to single-recirculation loop operation.

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 proposed license amendment establishes a revised safety limit MCPR value of 1.09 [two-recirculation loop and 1.10 for single-recirculation loop operation] for use during Unit 1 Cycle 12 operation. General Electric (GE) has determined that both generic and plant-specific evaluations [two-loop operation] yield the same calculated safety limit MCPR value. Additionally, a document referenced by the Technical Specification 6.9.3.2 of methodologies used in determining core operating limits is being removed.

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 have been established, consistent with NRC[-] approved methods, to ensure that fuel performance during normal, transient, and accident conditions is acceptable.

The probability of an evaluated accident is not increased by revising the safety limit MCPR value to 1.09 [two-loop/1.10 single-loop]. The change does not require any physical plant modifications or physically affect any plant components. Therefore, no individual precursors of an accident are affected.

The proposed license amendment establishes a revised safety limit MCPR that ensures the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences. Specifically, the reload analysis demonstrates that a safety limit MCPR value of 1.09 [two-loop/1.10 single-loop] ensures that less than 0.1 percent of the fuel rods will experience boiling transition during any plant operation if the limit is not violated.

The methods for calculating the safety limit MCPR have been approved by the NRC and are described in GE's reload licensing methodology topical report NEDE–24011, "General Electric Standard Application for Reactor Fuel (GESTAR II)." Based on (1) the determination of the new safety limit MCPR value using conservative approved methods, and (2) the operability of plant systems designed to mitigate the consequences of accidents not having been changed; the consequences of an accident previously evaluated have not been increased.

Additionally, removal of the footnote on the safety limit MCPR value in Technical Specification 2.1.2 and removal of reference "c" from the document list in Technical Specification 6.9.3.2 will not increase the probability or consequences of accidents previously evaluated. The footnote on the safety limit MCPR value in Technical Specification 2.1.2 and reference "c" in Technical Specification 6.9.3.2 were associated with the safety limit MCPR value of 1.10 for Unit 1 Cycle 11 operation. Since the current safety limit MCPR value of 1.10 applies only to Unit 1 Cycle 11 operation, the footnote on the safety limit MCPR value in Technical Specification 2.1.2 and the reference "c" in Technical Specification 6.9.3.2 are no longer needed and should be deleted. Thus, removal of the footnote on the safety limit MCPR value in Technical Specification 2.1.2 and removal of reference "c" from Technical Specification 6.9.3.2 is an administrative change that has no effect on the probability or consequences of accidents previously evaluated.

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

This proposed license amendment involves a revision of the safety limit MCPR from 1.10 to 1.09 [two-loop/1.10 single-loop] based on the results of both cycle-specific and generic analyses, removal of the footnote on the safety limit MCPR value in Technical Specification 2.1.2, and the removal of a document reference listed in Technical Specification 6.9.3.2 describing the methods used only during Unit 1 Cycle 11 to determine core operating limits. 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. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable modes of operation. Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

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

As previously stated, the methods for calculating the safety limit MCPR have been previously approved by the NRC and are described in GE's reload licensing methodology topical report NEDE–24011. Use of these methods ensures that the resulting safety limit MCPR satisfies the fuel design safety criteria that less than 0.1 percent of the fuel rods experience boiling transition if the safety limit is not violated. Based on the assurance that the fuel design safety criteria will be met, the proposed license amendment 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: 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: Pao-Tsin Kuo (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: May 16, 1997.

Description of amendment request: The proposed changes would replace the existing Technical Specification (TS) 4.6.2.3 a.2 cooling water flow rate of 1425 gpm with a new value of 1300 gpm.

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.

Cooling water flow to the Containment Fan Coolers is provided by the Emergency Service Water (ESW) System, and Emergency Service Water is not an initiating system in any FSAR [Final Safety Analysis Report] Chapter 15 analyses. Revising the minimum cooling water flow to the Containment Fan Coolers will not increase the probability of initiating any previously evaluated accident, because Containment Fan Cooler performance and integrity will not be adversely affected. The heat removal capacity of the Containment Fan Coolers will be maintained consistent with the assumptions used in the existing HNP [Harris Nuclear Plant] containment analyses, and, therefore, containment integrity should not be challenged.

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.

The proposed amendment will not create any new accident scenarios, because the change does not introduce any new single failures, adverse equipment or material interactions, or release paths.

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.

Although the proposed amendment replaces the TS 4.6.2.3 a.2 cooling water flow rate of 1425 gpm with a lower flow rate of 1300 gpm, a cooling water flow rate of greater than or equal to 1300 gpm maintains adequate heat removal capacity as required by existing HNP containment analyses. The Bases for TS 4.6.2.3 a.2 is to ensure that adequate heat removal capacity is available, when the Containment Fan Coolers are operated in conjunction with the Containment Spray Systems, during post-LOCA [Loss-of-Coolant Accident] conditions to prevent the pressure inside containment from exceeding its design rating.

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: Pao-Tsin Kuo (Acting).

Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit No. 2, Shippingport, Pennsylvania

Date of amendment request: October 22, 1997

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) by reducing the reactor coolant system (RCS) specific activity limits in accordance with Generic Letter 95–05. The definition of DOSE EQUIVALENT I–131 would be replaced with the Improved Standard TS definition wording in the first sentence and an equation added based on dose conversion factors derived from International Commission on Radiation Protection (ICRP) ICRP–30. TS 3.4.8, Specific Activity, would be revised by reducing the DOSE EQUIVALENT I–131 limit from 1.0 [micro] Ci[curies]/gram to 0.35 [micro]Ci[curies]/gram. Item 4.a in TS Table 4.4–12, Primary Coolant Specific Activity Sample and Analysis Program, TS Figure 3.4–1, and the Bases for TS 3/4.4.8 would be modified to reflect the reduced DOSE EQUIVALENT I–131 limit.

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

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

The proposed change reduces the reactor coolant system (RCS) specific activity limits of Specification 3.4.8 from 1.0 [micro]Ci/gram to 0.35 [micro]Ci/gram and lowers the graph in Figure 3.4–1 by 39 [micro]Ci/gram following the guidance provided in Generic Letter (GL) 95–05. This reduces the RCS acvitity allowed to leak to the secondary side when the plant is operating so that additional margin is available to support a higher allowable accident-induced leakage value as justified by analysis.

The proposed changes to Specification 3.4.8 and the definition of DOSE EQUIVALENT I–131 ensure these requirements are consistent with the latest analyses.

These changes implement the more restrictive RCS activity limits in accordance with applicable analyses and GL 95–05 to ensure the regulations are satisfied. Therefore, these changes do 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?

The proposed change does not alter the configuration of the plant or affect the operation with the reduced specific activity limit. By reducing the specific activity limit, the limit would be reached sooner to initiate evaluation of the out of limit condition. The proposed changes will not result in any additional challenges to the main steam system or the reactor coolant system pressure boundary. Consequently, no new failure modes are introduced as a result of the proposed changes. As a result, the main steam line break, steam generator tube rupture and loss of coolant accident analyses remain bounding. Therefore, the proposed change will 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 proposed change reduces the RCS specific activity limit to 0.35 [micro]Ci/gram along with lowering the Figure 3.4–1 limits by 39 [micro]Ci/gram. Reduction of the RCS specific activity limits allows an increase in the limit for the projected SG [steam generator] leakage following SG tube inspection and repair in accordance with the voltage-based SG tube alternate repair criteria (ARC). This follows the guidance provided in GL 95–05 and effectively takes margin available in the specific activity limits and applies it to the projected SG leakage for the ARC. This has been determined to be an acceptable means for accepting higher projected leakage rates while still meeting the applicable limits of 10 CFR [Part] 100 and GDC [General Design Criterion] 19 with respect to offsite and control room doses.

The capability for monitoring the specific activity and complying with the required actions remains unchanged. In addition, there is no resultant change in dose consequences. 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: 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.

Duke Energy Corporation, 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 3, 1998.

Description of amendment request: The licensee proposed to revise Section 6.2.3.2 of the units' Technical Specifications. Currently, this section prescribes that the Catawba Safety Review Group (SRG) be composed of at least five individuals and at least three of these shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his/her field, at least 1 year of which experience shall be in the nuclear field. The licensee proposed to revise this section to provide the option of replacing one of the three degreed individuals with one with at least 15 years of professional level experience in his/her field, at least 10 years of which experience shall be in the nuclear field, at least 3 years of which nuclear experience shall be supervisory/managerial experience in engineering, and shall hold or have held a Senior Reactor Operator license. The licensee also proposed to editorially revise this section.

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 analysis is presented below.

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

No. The proposed amendment would only change administrative requirements related to personnel qualifications for one of the five SRG [Safety Review Group] positions. The SRG is an oversight group, and the individual who meets the new qualification requirements would be expected to perform at the same level of quality as an individual who meets the current qualification requirements. Changing qualification requirements for an individual who primarily performs an oversight function will not have any direct effect on the design or operation of any plant structures, systems, or components. No previously analyzed accidents were initiated by the functions of the SRG, and the SRG was not a factor in the consequences of previously analyzed accidents. Therefore, the proposed change would have no impact on the consequences or probabilities of any previously evaluated accidents.

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

No. The proposed change would not lead to any hardware or operating procedure change. Hence, no new equipment failure modes or accidents from those previously evaluated will be created.

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

No. Margin of safety is associated with confidence in the design and operation of the plant. The proposed change to the Technical Specifications does not involve any change to plant design or operation. Thus, the margin of safety previously analyzed and evaluated is maintained.

Based on this analysis, 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: York County Library, 138 East Black Street, Rock Hill, South Carolina.

Attorney for licensee: Mr. Paul R. Newton, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: February 7, 1997.

Description of amendment request: The proposed amendment, if approved, would revise Technical Specification (TS) as delineated below:

1. 4160 Volt Tie From Unit 2.

TS sections 3.7.2.b & d to delete reference to the optional use of the 4160 volt tie from the unit 2 transformer.

2. Emergency Load Sequence and Power Transfer.

a. The testing required by Section 4.5.1.1.b of the TS would be considered satisfactory if the pumps have started and valves have completed travel. The need to evidence the successful starting of pumps and fans and the complete travel of valves by observation of control board component operating lights will be deleted. Neither would a second means of verification, such as: the station computer or control board indicating lights initiated by separate limit switch contacts be required.

b. Section 4.5.1.2.b would be revised in the same manner as 4.5.1.1.b above.

3. Reactor Building Cooling and Isolation System.

a. Section 4.5.3.1.a.1 of the TS would be revised to delete the need to simultaneously test start a spray pump using a Reactor Building 30-psi high pressure test signal while testing the emergency loading sequence.

The proposed change also eliminates the need to evidence the successful starting of the spray pumps by observation of the control board indicating lights or the use of the station computer for Sections 4.5.3.1.a.1 and 4.5.3.1.b.2.

4. Instrument Surveillance Requirements.

Table 4.1–1 of the TS would be revised to delete the strong motion accelerometer and its quarterly battery check surveillance requirement.

5. Air Intake Tunnel (AIT) Fire Protection Systems.

Section 5.5 of the TS would be deleted. The description of the equipment contained in Section 5.5 would be transferred to the Final Safety Analysis Report (FSAR).

6. Hydrogen Recombiner System.

The Bases for Section 4.4.4 TS would be changed to reflect a reduction in the time interval for operation of the hydrogen recombiner following a loss of cooling accident (LOCA) from 9.8 to 9 days.

7. Various editorial and typographical errors would be corrected.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The revised TS eliminate overly prescriptive requirements for evidencing component performance, the requirement for redundant diesel block loading tests,

instrumentation from SR [surveillance requirement] tables having no associated LCO [limiting condition for operation], AIT fire protection systems descriptive text, and correct previous typographical errors. Several of the proposed revisions involve changes which are consistent with NUREG–1430, the Revised Standard Technical Specifications (RSTS) for B&W plants. The reliability of systems and components depended upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed changes because assurance of system and equipment availability is maintained by surveillance testing program requirements.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The revised surveillance requirements create no new failure modes. Verification of equipment operation continues to be required by plant procedures. Elimination of the AIT fire protection system descriptive text from the TSs would not create a new or different kind of accident since the change has no effect on surveillance methodology and frequency requirements. They are maintained in the Fire Protection Program.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety because no operating limits are affected.

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: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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

NRC Project Director: Cecil O. Thomas, Director.

Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: March 3, 1998.

Description of amendment request: The proposed revision to the Millstone Unit 3 licensing basis would eliminate the requirement to have the recirculation spray system directly inject into the reactor coolant system following a design basis accident.

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:

Northeast Nuclear Energy Company (NNECO) has reviewed the proposed revision in accordance with 10CFR50.92 and has concluded that the revision does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not satisfied. The proposed revision does not involve an SHC because the revision would not:

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

The change to the Emergency Operating Procedures (EOP) to eliminate the use of Recirculation Spray System (RSS) direct injection during cold and hot leg recirculation does not effect the probability of any accident. The elimination of the requirement to have RSS directly [inject] into the reactor coolant system did not increase the consequences of the previously evaluated accidents. These consequences were evaluated based on very conservative assumptions concerning the containment pressure after the design basis Loss of Coolant Accident (LOCA), containment integrated leakage rates, and the fraction of the sprayed volume. None of these assumptions were affected by the elimination of the direct cold-leg injection.

Therefore, the proposed revision does not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The modification to the RSS did not create the possibility of a new or different accident from those previously analyzed. The change involved elimination of the direct injection flow path from the design basis of the system but did not involve physical modifications to the system itself. The operability of the affected valves within the direct injection alignments remained unchanged and these paths were still available to the operators for contingencies beyond the design basis. The EOPs provided clear and explicit guidance to that effect.

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

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

In considering the impact on the margin of safety as defined in the bases of the Technical Specifications, the impact of the change on the design basis analysis of the fission product barriers must be evaluated.

The minimum Emergency Core Cooling System flow requirement for long-term core cooling is that the modified alignment deliver sufficient flow to satisfy the inventory lost to the boil off in the vessel due to the decay heat and the extended boiling from hot metal in the downcomer and the lower plenum. The analysis determined that these requirements were being met.

The elimination of the direct injection resulted in a flow reduction through the RSS heat exchanger, from approximately 4000 gpm [gallons per minute] to 1200 gpm, thus reducing the rate of the heat transfer from the containment to the service water system. The design basis of the containment heat removal systems (circa 1986) is that the containment pressure will decrease to subatmospheric within one hour after the Design Basis Accident to compensate for the reduction in heat removal from the containment, a smaller allowable RSS pump degradation was assumed in the revised containment analysis. The original RSS pump performance curve was based on a 10 percent reduction in developed head from the design curve. For the modification, a 5 percent reduction was used. The results of the analysis show that with these changes the design basis of maintaining subatmospheric containment pressure was met.

Based on the above, elimination of the direct injection did not reduce the margin of safety because there was no violation of the acceptance limits and no weakening of the protective boundaries.

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

In conclusion, based on the information provided, it is determined that the proposed revision does not involve an SHC.

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, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

NRC Deputy Director: Phillip F. McKee.

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

Date of amendment requests: November 2, 1995, as supplemented by letter dated January 9, 1998. The January 9, 1998, submittal supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that was published on April 10, 1996 (61 FR 15995).

Description of amendment requests: In the November 2, 1995, letter, the licensee proposed to revise Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' to extend the offsite circuit completion time and to extend the allowed outage time for an emergency diesel generator. The January 9, 1998, letter modifies the original request to (1) further extend the offsite completion time and allowed outage time for an emergency diesel generator, and (2) add a new TS 5.5.2.14, ''Configuration Risk Management Program,'' that ensures a proceduralized probabilistic risk assessment-informed process is in place that assesses the overall impact of plant maintenance on plant risk.

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 Emergency Diesel Generators (EDGs) are backup alternating current power sources design to power essential safety systems in the event of a loss of offsite power. EDGs are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The EDGs provide backup power to components that mitigate the consequences of accidents. The proposed changes to the Completion Times do not affect any of the assumptions used in the deterministic safety analysis.

To fully evaluate the effect of the EDG Completion Time extension, Probabilistic Safety Analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of accidents previously evaluated.

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.8.1 does not affect the probability or the consequences of an accident.

Therefore, this change does not involve a significant increase in the probability or consequences of any 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.

This proposed change does not alter the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes do not affect the Limiting Conditions for Operation or their Bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes and these evaluations determined that the changes are either risk neutral or risk beneficial.

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

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

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

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

NRC Project Director: William H. Bateman.

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

Date of amendment requests: December 19, 1997.

Description of amendment requests: The licensee proposed to revise Technical Specification (TS) 3.4.9, ''Pressurizer,'' to reduce the allowable pressurizer water volume for pressurizer operability.

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 limiting events impacted by this Technical Specification change have been reanalyzed. These events are the Chemical and Volume Control System (CVCS) Malfunction and CVCS Malfunction With a Concurrent Single Failure of an Active Component, Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation (Including Single Failure of an Active Component), and Feedwater System Pipe Breaks. The probability of these events is not changed by the restriction of the pressurizer level to 57%. An operator action time of 15 minutes has been identified for the CVCS malfunction and inadvertent ECCS operation events. Based on the availability of operator alarms and indications and operator Simulator training, 15 minute operator action is sufficient to recognize and mitigate the inadvertent CVCS or ECCS operation. Therefore, this change will not involve an increase in the probability or consequences of any previously evaluated accident.

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

This amendment request does not involve any change to plant equipment or operation. All the events identified in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR) were evaluated to determine the impact of the change in pressurizer level. In addition to the normally analyzed Inadvertent Operation of the ECCS During Power Operation event a concurrent single failure of an active component was considered in this evaluation. The analysis of this event with single failure of an active component produced consequences that are bounded by the CVCS malfunction with single failure of an active component. No new or different kind of accident will be created as a result of this Technical Specification change. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This amendment request does not change the manner in which safety limits, limiting safety settings, or limiting conditions for operation are determined. There are no changes to the acceptance criteria for these events as a result of the proposed reduction in the maximum pressurizer water level. This change does not reduce a margin of safety since it lowers allowed pressurizer operational level to 57%. An operator action time of 15 minutes has been identified for the CVCS malfunction and inadvertent ECCS operation events. Based on the availability of operator alarms and indications, and demonstrated operator response in Simulator training, 15 minute operator action has been demonstrated to be adequate to recognize and mitigate the inadvertent CVCS or ECCS operation. Therefore, this proposed change does not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Main Library, University of California, Irvine, California 92713.

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

NRC Project Director: William H. Bateman.

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

Date of amendment requests: January 2, 1998.

Description of amendment requests: The licensee proposed to revise Technical Specification (TS) 3.7.5, ''Auxiliary Feedwater (AFW) System,'' to indicate the turbine driven AFW pump is operable when running in the manual mode to support plant startups, shutdowns, and testing.

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

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

Probabilistic analyses have been performed in support of declaring P140 operable when the pump is manually actuated and operating.

The results show that, considering P–140 to be in test for an entire year, the core damage risk of a Main Steam Line Break/Feedwater Line Break (MSLB/FWLB) slightly increases (4.3E–8/yr) while the risk due to other initiating events decreases (3E–7/yr). The net core damage impact of P–140 in test for an entire year is a Core Damage Frequency (CDF) decrease of 2E–7/yr. Having P140 operating instead of being in standby increases its reliability. This increased reliability reduces the risk due to other initiating events, such as loss of main feedwater, medium and small Loss of Coolant Accidents (LOCAs), Steam Generator Tube Rupture (SGTR), and Loss of Offsite Power (LOP), which require Auxiliary Feedwater (AFW) and which occur with much greater frequency than MSLB/FWLB. With the overall CDF reduction a result of considering P140 being in a test configuration for an entire year, the actual cumulative risk incurred is the weighted fraction that P140 is in the test configuration over a year period. Based on past experience, the pump is running in manual approximately 500 minutes/year, which results in an annual net cumulative CDF reduction on the order of 2E–10/yr due to running P140 in the manual mode.

Therefore, the operation of the facility in accordance with this 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.

This change does not involve a plant hardware modification or allow the operation of any plant equipment in any way other than originally designed. This change only affects the administrative tracking of the turbine-driven AFW pump when the steam driven AFW pump is operating in the manual mode.

Therefore, the operation of the facility in accordance with this proposed change 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.

Pump history shows the pump is run approximately 500 minutes per year. In all cases except for the one postulated scenario of the Main Steam Isolation Signal followed by an Emergency Feedwater Actuation Signal the turbine-driven AFW pump is not susceptible to being tripped. Also, this postulated scenario does not affect the capability of the motor-driven AFW pumps.

Even though there is a small increase in the CDF from the AFW steam driven pump operating in manual mode based on the possibility of a MSLB/FWLB, also considering other initiating events results in an annual net cumulative CDF reduction on the order of 2E–10/yr due to P140 running in the manual mode.

Therefore, the operation of the facility in accordance with this 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 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: Main Library, University of California, Irvine, California 92713.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: August 20, 1997, as supplemented by letters dated September 18, 1997 and October 31, 1997.

Description of amendment request: The proposed change would revise the Vermont Yankee Technical Specifications Section 6.0, ''Administrative Controls,'' to add and revise reference to NRC-approved methodologies which will be used to generate the cycle-specific thermal operating limits in the Vermont Yankee Core Operating Limits Report.

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

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

The change updates the Technical Specifications to include an NRC approved method reference to allow calculation of thermal limits with a revised method. It does not affect plant operation and will not weaken or degrade the facility.

2. The proposed change will not create the possibility of a new or different kind of accident since the change is administrative. No physical alterations of the plant, setpoint changes, or operating conditions are proposed.

3. The proposed change will not involve a significant reduction in a margin of safety. The change involves an update to the Administrative Controls in Section 6.0 of the Technical Specifications by adding a reference to NRC approved methods. This administrative change does not alter plant safety margins.

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: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037–1128.

NRC Project Director: Cecil O. Thomas, Director.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: February 4, 1998.

Description of amendment request: The amendment would revise Technical Specification 3.2.4, quadrant power tilt ratio (QPTR), and associated Bases, to clarify the required actions for the limiting condition for operation (LCO) and other changes consistent with the technical specification conversion application submitted by letter dated May 15, 1997.

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.

1. Requirements for Determining QPTR

The Action to calculate QPTR once per hour until THERMAL POWER was reduced to less than 50% RATED THERMAL POWER (RTP) when QPTR exceeds the LCO requirements would be deleted and replaced by a new requirement to determine QPTR at least once per 12 hours.

The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. The frequency with which QPTR is calculated is not assumed in the initiating events for any accident previously evaluated. In addition, the change does not involve any new operating activities or hardware change. Therefore, the proposed change would not significantly increase the probability of an accident previously evaluated.

Once THERMAL POWER has been reduced appropriately in proportion to the amount that QPTR exceeds 1.00, any additional change would be sufficiently slow that a 12-hour interval for recalculating QPTR will provide an adequate level of protection. Therefore, the proposed change will not significantly increase the consequences of any accident previously evaluated.

2. Completion Time for Resetting the Power Range Neutron Flux-High Trip Setpoints

The proposed change to allow 72 hours for resetting the Power Range Neutron Flux-High trip setpoints involves only the compensatory measures to be taken should the QPTR be outside its limit. These compensatory measures are not assumed in the initiating events for any accident previously evaluated. The proposed actions recognize that the required reduction in power (3% for each 1% of indicated QPTR in excess of 1.00) provide adequate margin for fuel design limits so that consequences of assumed accidents would not be significantly affected. Therefore, the proposed change will not adversely affect the probability or consequences of any accident previously evaluated. Further, by permitting more time to perform resetting the trip setpoints, the chances of a transient may be reduced.

3. Delete(tion) of the Actions (a.3., a.4.) for verifying QPTR to be restored within 24 hours and for identifying and correcting the cause of the out-of-limit condition prior to increasing THERMAL POWER

The proposed changes would delete current Actions a.3. and a.4. and add new Actions for QPTR out of limit including requirements for measuring $F_Q(Z)$ and F^N delta H prior to and following a return to power and performing safety analyses to verify safety requirements are met prior to increasing power above the limits of Action a.1. The proposed changes involve only the compensatory measures to be taken should the QPTR be outside its limit. These compensatory measures are not assumed in the initiating events for any accident previously evaluated. Therefore, the proposed change will not affect the probability or consequences of any accident previously evaluated.

4. Deletion of the Actions for QPTR in excess of 1.09

The proposed change would delete the required Actions for QPTR in excess of 1.09 and Actions for QPTR in excess of 1.02 are followed for all instances where QPTR exceeds 1.02. The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. These compensatory measures are not assumed in the initiating events for any accident previously evaluated. The proposed actions recognize that the required reduction in power (3% for each 1% of indicated QPTR in excess of 1.00) provide adequate margin for fuel design limits so that consequences of assumed accidents would not be significantly affected. Therefore, the proposed change will not affect the probability or consequences of any accident previously analyzed.

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

1. Requirements for Determining QPTR

The proposed change for calculating QPTR once every 12 hours does not involve a physical alteration to the plant or change the method by which any safety-related system performs its function. The manner in which the plant would be operated would not be altered. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

2. Completion Time for Resetting the Power Range Neutron Flux-High Trip Setpoints

The proposed change to allow 72 hours for resetting the Power Range Neutron Flux-High trip setpoints does not involve a permanent physical alteration to the plant; no new or different kinds of equipment will be installed. The change would not alter the manner in which the plant would be operated only the timing of actions that provide potential mitigation of accidents. Thus, the change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Delete the Actions (a.3., a.4.) for verifying QPTR to be restored within 24 hours and for identifying and correcting the cause of the out-of-limit condition prior to increasing THERMAL POWER

The proposed changes would delete current Actions a.3, and a.4. and add new Actions for QPTR out-of-limit including requirements for measuring $F_Q(Z)$ and F^N delta H prior to and following a return to power and performing safety analyses to verify safety requirements are met prior to increasing power above the limits of Action a.1. The proposed changes do not involve a physical alteration to the plant; no new or different kinds of equipment would be installed. The changes would not alter the manner in which the plant would be operated only the timing of actions that provide potential mitigation of accidents. Thus, the changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

4. Deletion of the Actions for QPTR in excess of 1.09

The proposed change would delete the required Actions for QPTR in excess of 1.09 and Actions for QPTR in excess of 1.02 are followed for all instances where QPTR exceeds 1.02. The proposed change does not involve a physical alteration to the plant or changes in the way in which the plant is operated. The proposed change involves only the compensatory measures to be taken should QPTR be outside its limit. The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed alternate methods of dealing with QPTRs in excess of 1.09. Therefore, there is no 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.

1. Requirements for Determining QPTR

The proposed change for calculating QPTR once every 12 hours does not change any accident analysis assumptions, initial conditions or results. The proposed change will continue to ensure that the plant is maintained in a safe condition while QPTR is in excess of its limit. Additionally, calculating QPTR once per 12 hours as opposed to every hour while QPTR is in excess of its limit would avoid the diversion of personnel resources from corrective actions with regard to meeting the LCO. Therefore, the proposed change will not involve a significant reduction in any margin of safety.

2. Completion Time for Resetting the Power Range Neutron Flux-High Trip Setpoints

The proposed change to allow 72 hours for resetting the Power Range Neutron Flux-High trip setpoints will continue to ensure that the plant is maintained in a safe condition within the envelope of the safety analyses while QPTR is in excess of its limit. The proposed actions recognize that the required reduction in power (3% for each 1% of indicated QPTR in excess of 1.00) provide adequate margin for fuel design limits so that consequences of assumed accidents would not be significantly affected. Therefore, the proposed change will not involve a significant reduction in any margin of safety.

3. Delete the Actions (a.3., a.4.) for verifying QPTR to be restored within 24 hours and for identifying and correcting the cause of the out-of-limit condition prior to increasing THERMAL POWER

The proposed changes would delete current Actions a.3. and a.4 and add new Actions for QPTR out-of-limit including requirements for measuring $F_Q(Z)$ and F^N delta H prior to and following a return to power and performing safety analyses to verify safety requirements are met prior to increasing power above the limits of Action a.1. The proposed changes will continue to ensure that the plant is maintained in a safe condition within the envelope of the safety analysis while QPTR is in excess of its limit. Therefore, the proposed changes will not involve a significant reduction in any margin of safety.

4. Deletion of the Actions for QPTR in excess of 1.09

The proposed change would delete the required Actions for QPTR in excess of 1.09

and Action for QPTR in excess of 1.02 are followed for all instances where QPTR exceeds 1.02. The proposed change will continue to ensure that the plant is maintained in a safe condition within the envelope of the safety analyses while QPTR is in excess of its limit. While different actions are taken in response to a QPTR in excess of 1.09, the proposed change will assure that accident analyses assumptions continue to be met. Therefore, the proposed changes will not involve a significant reduction in any 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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: February 4, 1998.

Description of amendment request: The amendment would revise the technical specifications to (1) create separate functional units for the analog and digital portions of the engineered safety features actuation system (ESFAS) function associated with starting the turbine-driven auxiliary feedwater pump on a loss of offsite power, and (2) add a table notation to clarify that the testing of the time delay relays for the 4 kV undervoltage, loss of voltage and grid degraded voltage portion of the ESFAS is performed as part of the channel calibration.

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The recognition that different OPERABILITY and surveillance requirements apply to analog vs. digital circuitry does not impact any previously analyzed accidents. The clarification that testing of the time delay relays is performed as part of the CHANNEL CALIBRATION does not impact any previously analyzed events. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change does not alter the current method or procedures for meeting the surveillance requirements in Table 4.3–2. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. 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.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The separation of analog and digital portions of Functional Unit 6.f or the clarification of testing of the time delay relays will not impact the normal method of plant operation.

The OPERABILITY requirements, ACTION Statement, and surveillance requirements for the analog portion, new Functional Unit 6.f.1), are identical to those of Functional Unit 8.a, while the requirements for the digital portion, new Functional Unit 6.f.2), are consistent with the current technical specifications, other than the new ACTION Statement 30 provisions that defer to the TDAFW pump Specification 3.7.1.2 requirements and the performance of a TADOT during appropriate plant conditions. These changes do not change any ESFAS design standard and are appropriate for digital functions such as this.

Testing of the time delay relays has been performed as part of the 18 month CHANNEL CALIBRATION. The tolerancesfor the time delay relays are sufficient to account for relay drift encountered during the 18 month surveillance testing. The calculated tolerances for the time delay setpoints have been evaluated to insure that safety-related systems, subsystems and components would not be adversely affect[ed] by the drift within the permissible tolerance band.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any 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 locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Project Director: William H. Bateman.

Previously Published Notices of Consideration of Issuance of Amendments toFacility 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.

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

Date of amendment request: January 26, 1998.

Brief description of amendment request: The proposed amendment would change the SSES Technical Specifications facility staff requirements to allow an individual who does not hold a current senior reactor operator (SRO) license to hold the position of Manager-Nuclear Operations (MNO) and require an individual serving in the capacity of the Operations Supervisor-Nuclear to hold a current SRO license and report directly to the MNO and be responsible for directing the licensed activities of licensed operators.

Date of publication of individual notice in Federal Register: February 24, 1998 (63 FR 9270).

Expiration date of individual notice: March 26, 1998.

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

TU Electric Company, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: February 25, 1998, TXX–98050.

Description of amendment request: The proposed amendment would be a temporary change to the Technical Specifications to remove the requirement to demonstrate the load shedding feature of MCC XEB4–3 as part of Surveillance Requirements (SRs) 4.8.1.1.2f.4)a) and 4.8.1.1.2f.6)a) until the plant startup subsequent to the next refueling outage or until an outage of greater than 24 hours in duration for each respective unit. This temporary change is requested as a result of the failure to confirm the load shedding feature of MCC XEB4–3 during the last performance of these SRs for the Unit 1 and Unit 2 train B diesel generators (DGs).

Date of individual notice in the **Federal Register:** March 9, 1998, (63 FR 11458).

Expiration date of individual notice: April 8, 1998.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

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 Nos. 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: March 18, 1997, as supplemented by letters dated July 28, 1997, and September 9, 1997.

Brief description of amendments: The amendments revise the operating licenses to reflect approval of Amendment 42 to the Palo Verde Nuclear Generating Station Physical Security Plan. The amendments revise the methods used to search materials, packages, and personnel prior to their entry into the protected area, as described in the security plan.

Date of issuance: March 4, 1998.

Effective date: March 4, 1998.

Amendment No.: Unit 1–115; Unit 2–108; Unit 3–87.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the operating licenses.

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52580).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 4, 1998.

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, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: December 17, 1997, as supplemented by letters dated February 6, 1998 and March 12, 1998.

Brief description of amendment: The proposed change would revise Technical Specifications Section 5.6.5, ''Core Operating Limits Report.'' The revisions add reference to an additional approved methodology for correlating departure from nucleate boiling (DNB) ratios. The added methodology is the Siemens Power Corporation Topical Report, EMF–92–153(P)(A), ''HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel.''

Date of issuance: March 16, 1998.

Effective date: March 16, 1998.

Amendment No. 178.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4309). The February 6 and March 12, 1998 submittals provided clarifying information that did not affect the initial determination of no significant hazards considerations. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, 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: October 27, 1997.

Brief description of amendments: The amendments would change the Dresden and Quad Cities Technical Specifications (TS) to clarify the applicability, action and surveillance requirements for the Standby Liquid Control System (SLCS). The changes would make the current TS requirements for the SLCS consistent with the Improved Standard Technical Specifications (ISTS) contained in NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4.''

Date of issuance: March 6, 1998.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 167, 162, and 180, 178.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2277).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Consolidated Edison Company of New York, Docket No. 50–247, Indian PointNuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: October 2, 1996, as supplemented July 31, 1997.

Brief description of amendment: The amendment revises Figures 3.1.A–1, 3.1.A–2 and 3.1.A–3, Section 3.1.B and its Bases, Figures 3.1.B–1 and 3.1.B–2, and the Bases of Section 4.3 and Figure 4.3–1 of the Technical Specifications to incorporate the revised Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation.

Date of issuance: February 27, 1998.

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

Amendment No.: 195.

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

Date of initial notice in **Federal Register:** November 19, 1996 (61 FR 58901).

The July 31, 1997, letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 1998.

No significant hazards consideration comments received: No.

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

Duke Energy Corporation, 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: December 17, 1997Brief description of amendments: The amendments revise Section 6.9.1.9 of the Technical Specifications to reference updated or recently approved topical reports, which contain methodologies used to calculate cycle-specific limits contained in the Core Operating Limits Report.

Date of issuance: March 2, 1998.

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

Amendment Nos.: Unit 1–163; Unit 2–155.

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

Date of initial notice in **Federal Register:** January 28, 1998 (63 FR 4310).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 2, 1998.

No significant hazards consideration comments received: No.

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

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: July 21, 1997, as supplemented February 18, 1998.

Brief description of amendment: Technical Specification Change Request concerning Emergency Feedwater Surveillance Testing. This request is to make several changes to the ANO–2 Technical Specifications including extension of the emergency feedwater (EFW) pump surveillance testing frequency, a reduction in the minimum steam generator pressure required to perform the surveillance testing on the turbine-driven EFW pump, and a modification to the EFW pump testing requirements.

Date of issuance: March 12, 1998.

Effective date: March 12, 1998.

Amendment No.: 188.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications/license.

Date of initial notice in **Federal Register:** August 13, 1997 (62 FR 43367).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: September 23, 1997, as supplemented by letters dated February 27 and March 4, 1998.

Brief description of amendment: The amendment changes the Reactor Protective System (RPS) and Engineering Safety Actuation System (ESFAS) trip set point and allowable values for steam generator low pressure. The amendment also relocates the RPS and ESFAS response time tables from the Technical Specifications to the Safety Analysis Report as described in NRC Generic Letter 93–08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' dated December 29, 1993.

Date of issuance: March 12, 1998.

Effective date: March 12, 1998.

Amendment No.: 189.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications/license.

Date of initial notice in **Federal Register:** January 28, 1998, (63 FR 4311).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: September 23, 1997, as supplemented by letters dated February 27 and March 4, 1998.

Brief description of amendment: The amendment reduces the minimum required reactor coolant system flow rate in TS 3.2.5 until the ANO–2 steam generators are replaced. The reduced reactor coolant system flow requirement will account for plugging of up to approximately 30 percent of the tubes in the existing steam generators at ANO–2.

Date of issuance: March 12, 1998.

Effective date: March 12, 1998.

Amendment No.: 190.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications/license.

Date of initial notice in **Federal Register:** January 28, 1998, (63 FR 4312).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

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

Date of application for amendment: December 5, 1997, as supplemented December 11, 1997, January 9, February 12 and 19, 1998.

Brief description of amendment: To revise the Final Safety Analysis Report (FSAR) and the Improved Technical Specification Bases to reflect the modified reactor building fan recirculation system fan cooler starting logic.

Date of issuance: March 9, 1998.

Effective date: March 9, 1998.

Amendment No.: 165.

Facility Operating License No. DPR–31: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 15, 1998 (63 FR 2423). The supplemental letters dated December 11, 1997, January 9, February 12 and 19, 1998, 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 9, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida.

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P.O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Project Director: Frederick J. Hebdon.

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

Date of application for amendments: May 14, 1997, as supplemented by letter dated October 9, 1997 (published in **Federal Register** as May 15, 1997).

Brief description of amendments: The amendments revised the combined Technical Specifications (TS) for the Diablo Canyon Power Plant (DCPP) Unit Nos. 1 and 2 to revise the surveillance frequencies from at least once every 18 months to at least once per refueling interval (nominally 24 months) including (1) reactor coolant system total flow rate, (2) instrumentation for radiation monitoring, (3) instrumentation and controls for remote shutdown, (4) instrumentation for accident monitoring, and (5) several miscellaneous TS.

Date of issuance: February 27, 1998.

Effective date: February 27, 1998, to be implemented within 90 days of the date of issuance.

Amendment Nos.: Unit 1–123; Unit 2–121.

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

Date of initial notice in **Federal Register:** July 30, 1997 (62 FR 40855).

The October 9, 1997, supplemental letter provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 1998.

No significant hazards consideration comments received: No.

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

Power Authority of the State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: January 2, 1997, as supplemented November 13, 1997.

Brief description of amendment: The amendment changes the Technical Specifications by extending the surveillance interval for the functional testing of certain Inservice Inspection American Society of Mechanical Engineers Code Class 1, 2, and 3 pumps and valves from once a month to once a quarter.

Date of issuance: March 2, 1998.

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

Amendment No.: 178.

Facility Operating License No. DPR–64: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** March 26, 1997 (62 FR 14468).

The November 13, 1997, submittal contained clarifying information that did not change the staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

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

Date of application for amendment: December 14, 1995, as supplemented September 26, 1997.

Brief description of amendment: The amendment changes the James A. FitzPatrick Technical Specifications (TSs) to incorporate the inservice testing requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The amendment supplements Amendment No. 241, dated December 2, 1997, by issuing seven TS pages inadvertently omitted from Amendment No. 241.

Date of issuance: February 27, 1998.

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

Amendment No.: 242.

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

Date of initial notice in **Federal Register**: January 22, 1996 (61 FR 1635).

The September 26, 1997, letter 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 February 27, 1998.

No significant hazards consideration comments received: No.

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

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: December 15, 1997.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) to adopt Option B, of 10 CFR Part 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' to implement a performance-based approach for Type B and C testing. Additionally, the wording in the TSs would be modified for the previous adoption of Option B on Type A testing and a section added on the primary containment leakage rate testing program.

Date of issuance: February 27, 1998.

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

Amendment Nos: 207 and 188.

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

Date of initial notice in **Federal Register**: January 14, 1998 (63 FR 2281).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 1998.

No significant hazards consideration comments received: No.

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

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: December 17, 1997.

Brief description of amendments: The amendments extended the surveillance interval of the containment spray nozzle air flow test to ten years from five years.

Date of issuance: March 11, 1998.

Effective date: March 11, 1998.

Amendment Nos.: Unit 1—Amendment No. 94; Unit 2—Amendment No. 81.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4325).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 11, 1998.

No significant hazards consideration comments received: No.

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

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440 Perry Nuclear Power Plant, Unit 1, Lake County, Ohio.

Date of application for amendment: December 23, 1997.

Brief description of amendment: This amendment revised Technical Specification 3.8.1, ''A.C. Sources—Operating,'' consistent with the recommendations in NRC Generic Letter 94–01, ''Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators.''

Date of issuance: March 12, 1998.

Effective date: March 12, 1998.

Amendment No.: 92.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 28, 1998 (63 FR 4326).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

Date of application for amendment: July 11, 1997, as supplemented November 21, December 22, 1997, and February 6, 1998.

Brief description of amendment: The amendment revised Technical Specifications 3.7/4.7 and their associated Bases to incorporate Option B of Appendix J to 10 CFR 50, and editorial changes to TS Table 4.7.2

Date of Issuance: February 26, 1998.

Effective date: February 26, 1998, with full implementation within 30 days.

Amendment No.: 152.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: (62 FR 45465). The November 21, December 22, 1997, and February 6, 1998, letters did not change the initial proposed no significant hazards determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 26, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: November 20, 1997.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.10 and its associated Bases to eliminate the use of battery charger AB for meeting the requirement of the TS.

Date of issuance: March 5, 1998.

Effective Date: This license amendment is effective as of its date of issuance, to be implemented within 30 days.

Amendment No.: 153

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 31, 1997 (62 FR 68319).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 5, 1998.

No significant hazards consideration comments received: No.

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

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

Date of application for amendment: August 22, 1997, as supplemented by letter dated September 18 and October 31, 1997.

Brief description of amendment: The amendment revises the Technical Specifications to address the new low pressure CO2 suppression system for the East and West Switchgear Rooms and more clearly describes the separation of the two rooms.

Date of Issuance: March 6, 1998.

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

Amendment No.: 154.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 8, 1997 (62 FR 52590). Information provided by letter dated October 31, 1997, did not affect the original no significant hazards consideration.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated March 6, 1998.

No significant hazards consideration comments received: No.

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

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 April 24, 1998, 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 20555–0001, Attention: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. 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 the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

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

Date of application for amendments: February 5, 1998, as supplemented February 12, March 3 and 5, 1998.

Brief description of amendments: The amendments revised the surveillance requirements in Technical Specification (TS) 4.6.1.2 (Requirement a). The change to the referenced TS adds a footnote stating that the requirement for Type A testing will not apply to certain instrument line penetrations.

Date of issuance: March 10, 1998.

Effective date: Both units, as of the date of issuance.

Amendment Nos.: 173 and 146.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No. On February 5, 1998, the staff issued a Notice of Enforcement Discretion, which was immediately effective and remained in effect until this amendment was issued.

The Commission's related evaluation of the amendments, finding of emergency circumstances, consultation with the State of Pennsylvania, and final no significant hazards consideration determination are contained in a Safety Evaluation dated March 10, 1998.

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

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

NRC Project Director: John F. Stolz.

Dated at Rockville, Maryland, this 18th day of March 1998.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 98–7652 Filed 3–24–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Privacy Act of 1974, As Amended; Revisions to System of Records

AGENCY: Nuclear Regulatory Commission.

ACTION: System of records; proposed revisions.

SUMMARY: In accordance with the Privacy Act of 1974, as amended (Privacy Act), the Nuclear Regulatory Commission (NRC) is proposing to amend the notice describing the system of records (system) currently entitled NRC–32, ''Office of the Controller Financial Transactions and Debt Collection Management Records—NRC,'' by adding five new routine uses and revising five existing routine uses in order to permit NRC to comply with certain provisions of the Debt Collection Improvement Act of 1996 (DCIA), Public Law 104–134. The system notice was last published in the **Federal Register** on April 17, 1996.

EFFECTIVE DATE: The revised system of records will become effective without further notice on May 4, 1998, unless comments received on or before that date cause a contrary decision. If changes are made based on NRC's review of comments received, a new final notice will be published.

ADDRESSES: Send comments to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications staff. Hand deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. Federal workdays. Copies of comments received may be examined, or copied for a fee, at the NRC Public Document Room at 2120 L Street, NW., Lower Level, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Jona L. Souder, Freedom of Information Act/Privacy Act Section, Information Services Branch, Information Management Division, Office of the Chief Information Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–7170.

SUPPLEMENTARY INFORMATION:

The DCIA contains various provisions intended to maximize the collection of delinquent debts, minimize the costs of debt collection, reduce losses arising from debt management activities, rely on the experience and expertise of private sector professionals to provide debt collection services to Federal agencies, and ensure that the public is fully informed of the Federal government's debt collection policies and that debtors are cognizant of their financial obligations to repay amounts owed to the government and have all appropriate due process rights.

The proposed revisions to NRC–32 will permit NRC to implement several new techniques for collecting debts and claims authorized or required by the DCIA. New routine use I. will permit NRC to refer nontax debts over 180 days delinquent to the Department of the Treasury (Treasury) for administrative offset against payments due elsewhere in the government under the mandatory, government-wide Treasury Offset Program (TOP). TOP provides a single source for identifying delinquent debtors receiving government funds and, to the extent legally allowed, offsetting the delinquent debts using those same funds. New routine use m. will enable NRC to publicly disseminate the names of certain delinquent debtors and the existence of the debts for debt collection purposes. New routine use n. will enable NRC to match certain debtor records with the Department of Health and Human Services and the Department of Labor to obtain Taxpayer Identification Numbers required by the DCIA for each person doing business with Federal agencies. New routine uses o. and p. will permit NRC to disclose information if it decides or is required