(4) Minimize the burden of the collection of information on those who are to respond, including through the use of appropriate automated, electronic, mechanical, or other technological collection techniques or other forms of information technology, e.g., permitting electronic submission of responses.

Overview of this information collection:

(1) *Type of Information Collection:* *Extension of a currently approved information collection.*

(2) *Title of the Form/Collection:* Application for Waiver of Grounds of Excludability.

(3) *Agency form number, if any, and the applicable component of the Department of Justice sponsoring the collection:* Form I–690. Adjudications Division, Immigration and Naturalization Service.

(4) *Affected public who will be asked or required to respond, as well as a brief abstract:* Primary: Individuals or Households. This information on the application will be used by the Service in considering eligibility for legalization under sections 210 and 245A of the Immigration and Nationality Act.

(5) *An estimate of the total number of respondents and the amount of time estimated for an average respondent to respond:* 85 responses at 15 minutes (.25 hours) per response.

(6) *An estimate of the total public burden (in hours) associated with the collection:* 21 annual burden hours.

If you have additional comments, suggestions, or need a copy of the proposed information collection instrument with instructions, or additional information, please contact Richard A. Sloan, 202–514–3291, Director, Policy Directives and Instructions Branch, Immigration and Naturalization Service, U.S. Department of Justice, Room 4034, 425 I Street, NW., Washington, DC 20536. Additionally, comments and/or suggestions regarding the item(s) contained in this notice, especially regarding the estimated public burden and associated response time may also be directed to Mr. Richard A. Sloan.

If additional information is required contact Mr. Robert B. Briggs, Clearance Officer, United States Department of Justice, Information Management and Security Staff, Justice Management Division, 1331 Pennsylvania Avenue, NW., Suite 1220, Washington, DC 20530.

Dated: July 26, 2001.

Richard A. Sloan,

Department Clearance Officer, United States Department of Justice, Immigration and Naturalization Service.

[FR Doc. 01–19866 Filed 8–7–01; 8:45 am]

BILLING CODE 4410–10–M

DEPARTMENT OF LABOR

Office of the Secretary

Advisory Council on-Employee, Welfare and Pension Benefit Plans; Nominations for Vacancies

Section 512 of the Employee Retirement Income Security Act of 1974 (ERISA), 88 Stat. 895, 29 U.S.C. 1142, provides for the establishment of an ‘‘Advisory Council on Employee Welfare and Pension Benefit Plans’’ (the Council), which is to consist of 15 members to be appointed by the Secretary of Labor (the Secretary) as follows: Three representatives of employee organizations (at least one of whom shall be representative of an organization whose members are participants in a multi employer plan); three representations of employers (at least one of whom shall be representative of employers maintaining or contributing to multi employer plans); one representative each from the fields of insurance, corporate trust, actuarial counseling, investment counseling, investment management and accounting; and three representatives from the general public (one of whom shall be a person representing those receiving benefits from a pension plan). No more than eight members of the Council shall be members of the same political party.

Members shall be persons qualified to appraise the programs institute under ERISA. Appointments are for terms of three years. The prescribed duties of the Council are to advise the Secretary with respect to the carrying out of his or her function under ERISA, and to submit to the Secretary, or his or her designee, recommendations with respect thereto. The Council will meet at least four times each year, and recommendations of the Council to the Secretary will be included in the Secretary's annual report to the Congress ERISA.

The terms of five members of the Council expire on November 14, 2001. The groups or fields they represented are as follows: employee organizations, insurance, accounting, employers and the general public. The Department of Labor is committed to equal opportunity in the workplace and seeks a board-based and diverse ERISA Advisory Council membership.

Accordingly, notice is hereby given that any person or organization desiring to recommend one or more individuals for appointment to the ERISA Advisory Council on Employee Welfare and Pension Benefit Plans to represent any of the groups or fields specified in the preceding paragraph, may submit recommendations to Sharon Morrissey, Executive Secretary, ERISA Advisory Council, Frances Perkins Building, U.S. Department of Labor, 200 Constitution Avenue, NW., Suite N–5677, Washington, DC 20210. Recommendations must be delivered or mailed on or before October 1, 2001. Recommendations may be in the form of a letter, resolution or petition, signed by the person making the recommendation or, in the case of a recommendation by an organization, by an authorized representative of the organization.

Signed at Washington, DC, this 2nd day of August, 2001.

Ann L. Combs,

Assistant Secretary of Labor, Pension and Welfare Benefits Administration.

[FR Doc. 01–19868 Filed 8–7–01; 8:45 am]

BILLING CODE 4510–29–M

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 July 16, 2001 through July 27, 2001. The last biweekly notice was published on July 25, 2001 (66 FR 38756).

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 Administrative 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 7, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room). 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: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to pdr@nrc.gov.

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: May 24, 2001.

Description of amendment request: Generic Letter 96–04 informed all licensees of the issues concerning the use of Boraflex in spent fuel storage racks. In an October 15, 1996, response to the generic letter, the licensee stated that a reevaluation of the criticality analysis for the Oyster Creek fuel racks would be performed to consider Boraflex degradation including boron carbide loss. A reevaluation of the Oyster Creek criticality analysis including consideration of Boraflex degradation has been performed and the licensee is asking for review and approval of the proposed change to its licensing basis of spent fuel racks containing Boraflex.

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

The proposed amendment does not:

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

The accident of concern is a fuel bundle drop onto the top of a storage rack as described in DPR–16 License Amendment No. 76 dated September 17, 1984 and DPR–16 License Amendment No. 121. This accident was previously considered in an analysis that calculated the reactivity of two unpoisoned fuel assemblies separated only by water. The analysis shows a separation of 2.5 inches results in a reactivity k_∞ of 0.90. For a fuel assembly lying horizontally on the top of a rack, the separation distance would be $\simeq$ 14 inches. Since only water separation is considered and no credit is taken for Boraflex, there is no effect on this accident as described in the SAR [Safety Analysis Report].

The SAR identifies that k_{eff} for the spent fuel shall not exceed 0.95 accounting for uncertainties. This criticality analysis, which includes consideration of Boraflex degradation, shows the spent fuel pool K_{eff} will remain below 0.95 with a 95% probability at the 95% confidence level. Therefore, the revised criticality analysis for Boraflex degradation does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The change does not involve any plant systems associated with plant operation so safe plant operation will not be affected. This analysis does include a new consideration (dissolution of the Boraflex in the fuel racks) that had not been previously considered.

Nuclear safety is not effected [affected] since the required margin to criticality is maintained with consideration of Boraflex degradation. The current analysis uses the conservative assumption of coplanar gaps (i.e., all gaps occurring at the same axial plane). This is a very conservative assumption given gap measurement data at Oyster Creek and in the industry that shows an axial distribution of gaps.

The proposed criticality analysis utilizes an axial distribution of gaps. The analysis is based on the same fuel design and enrichment as the previous analysis, a GE7 8x8 fuel design having 4.0% enrichment and seven rods containing 3.0% Gd_3O_8 depleted to peak reactivity. The analysis assumes shrinkage up to 4.2% of panel length, gaps of 5.89 inches occurring in 75% of the panels, and 10% thinning (4 mils) of the panel thickness. The analysis conforms to regulatory and industry guidelines for criticality analyses and the calculated k_{eff} provides 95% probability at the 95% confidence level. The design limit is 0.9410 (5.0% design margin plus calculational uncertainity) and the spent fuel pool k_{eff} is 0.9381 including manufacturing uncertainties. This establishes the acceptability of the assumed Boraflex degradation against design limits.

The analysis, which includes the effect of Boraflex degradation, demonstrates that k_{eff} in the fuel racks remains below the license requirement of 0.95. The possibility of a new or different kind of accident from any accident previously evaluated is not created since k_{eff} remains below 0.95 when Boraflex degradation mechanisms are considered and the change does not involve any plant systems or procedures associated with plant operation.

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

As stated in Oyster Creek Technical Specification Section 5.3.1, the fuel pool k_{eff} is limited to 0.95 to assure [ensure] an ample margin to criticality. The new analysis demonstrates this margin is maintained given the Boraflex degradation assumed in the analysis that is based on industry and Oyster Creek specific observations and testing. The new analysis revises the Boraflex gap assumption to use a random axial distribution of gaps rather than a more conservative coplanar (gaps in same location in all fuel bundles) distribution. The axial distribution is more representative of actual gap locations observed at Oyster Creek (based on Blackness and BADGER testing) and other plants with similar rack designs. The assumption remains conservative since all Boraflex gaps are assumed to occur in the upper three-quarters of the rack height. This results in an over estimation of gaps in a smaller area that increases the reactivity penalty. Since the required k_{eff} limit of 0.95 is not exceeded, and the analysis remains conservative, this 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.

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard Correia, Acting.

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

Date of amendments request: June 15, 2001.

Description of amendments request: The proposed amendments delete requirements from the Technical Specifications (TSs) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated June 15, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Stephen Dembek.

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: July 17, 2001.

Description of amendment request: By letters dated October 4, 2000, and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Operating License and Technical Specifications (TS) to support steam generator replacement (SGR) and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). CP&L, in its letter of July 17, 2001, proposed to revise the Final Safety Analysis Report (FSAR) Chapter 15 accident analyses, which had been previously submitted as part of the October 4, 2000, SGR amendment request. The proposed revision to the accident analyses would adopt the alternate source term (AST) methodology, using the guidance of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below. The licensee's analysis is limited to its request to use the AST methodology for the accident analyses. The NRC staff's proposed no significant hazards consideration determination for the SGR amendment request, published in the **Federal Register** on November 1, 2000 (65 CFR 65338), and the Notice of Consideration of Issuance to Amendment to Facility Operating License and Opportunity for Hearing for the power uprate, published on February 6, 2001 (66 CFR 9110), remain valid for the other aspects of the SGR and power uprate amendment requests.

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

An alternative source term calculation has been performed for HNP which demonstrates that dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of the plant.

Therefore, the proposed amendment does not involve a significant 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 change does not affect plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change.

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

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

The proposed change is the implementation of the alternate source term methodology consistent with NRC Regulatory Guide 1.183. The proposed change does not significantly affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or FSAR. Accordingly, NRC Acceptance Limits are not significantly affected by this change.

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

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

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

NRC Section Chief: Patrick M. Madden, Acting.

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

Date of amendment request: July 16, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Sections 3.1.A, ''Reactor Coolant System Operational Components,'' 3.1.B, ''Reactor Coolant System [RCS] Heatup and Cooldown,'' 3.2, ''Chemical and Volume Control System,'' 3.3.A, ''Engineered Safety Feature Safety Injection and Residual Heat Removal Systems,'' and 4.3, ''Reactor Coolant System Integrity Testing,'' to incorporate revised reactor pressure vessel pressure-temperature limits to allow operation up to 25 effective full-power years (EFPY). The proposed amendment would also make changes to the associated TS Bases sections.

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 consequences of an accident previously evaluated.

The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the heatup and cooldown limitation curves. The proposed changes do not modify the RCS pressure boundary. That is, there are no changes in operating pressure, materials, or seismic loading. The proposed changes do not adversely affect the integrity of the RCS pressure boundary such that its function in the control of radiological consequences is affected. The proposed heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10CFR50 Appendix G, and ASME B&PV Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code], Section XI, Appendix G in conjunction with ASME Code Cases N–640 and N–588. The proposed heatup and cooldown limitation curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of RPV [reactor pressure vessel] beltline materials. Use of this methodology provides compliance with the intent of 10CFR50 Appendix G and provides margins of safety that ensure non-ductile failure of the RPV will not occur.

The proposed heatup and cooldown limitation curves prohibit operation in regions where it is possible for non-ductile failure of carbon and low alloy RCS materials to occur. Hence, the primary coolant pressure boundary integrity will be maintained throughout the limit of applicability of the curves, 25 EFPY. Operation within the proposed OPS [overpressure protection system] limits ensures that overpressurization of the RCS at low temperatures will not result in component stresses in excess of those allowed by the ASME B&PV Code Section XI Appendix G.

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

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 proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the heatup and cooldown limitation curves were generated in accordance with the fracture toughness requirements of 10CFR50 Appendix G and ASME B&PV Code, Section XI, Appendix G in conjunction with ASME Code Cases N–588 and N–640. Compliance with the heatup and cooldown limitation curves will ensure that conditions in which non-ductile failure of the RCS pressure boundary materials is possible will be avoided. Compliance with the proposed OPS limits will ensure that the RCS will be physically protected against overpressurization events during low temperature operation when the fracture toughness properties of the carbon and low alloy components are at their lowest.

No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the heatup and cooldown limitation curves and the OPS limits do not affect any activities or equipment other than the RCS pressure boundary and are not assumed in any analysis to initiate or mitigate any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

The proposed TS changes do not involve a significant reduction in the margin of safety.

The revised heatup and cooldown limitation curves and OPS limits provide more operating flexibility than the current heatup and cooldown limitation curves. Industry experience since the inception of pressure-temperature limits in the 1970s confirms that some of the original methodologies used to develop the heatup and cooldown limitation curves are overly conservative. Accordingly, ASME Code Cases N–588 and N–640 take advantage of the acquired knowledge by establishing more realistic methodologies for development of the heatup and cooldown limitation curves. Therefore, operational flexibility is gained and an acceptable margin of safety to reactor pressure vessel non-ductile type fracture is maintained.

The revised heatup and cooldown limitation curves and OPS limits are established in accordance with current regulations and the ASME B&PV Code 1996 version. These proposed changes are acceptable because the ASME B&PV Code maintains the margin of safety required by

10CFR50.55(a). Because operation will be within these limits, the RCS materials will continue to behave in a ductile manner consistent with the original design bases.

The proposed changes to the allowable operation of charging and safety injection pumps when OPS is required to be operable is consistent with the IP2 [Indian Point 2] licensing bases but implements the licensing bases in a more conservative manner than the current TS. The change in OPS surveillance frequency has been previously evaluated by the NRC to involve an insignificant increase in risk. That insignificant increase in risk is offset by the adverse effects of the alternatives of either 1.) delaying forced cooldowns until OPS testing is complete; 2.) complicating cooldown operations by imposition of limits required when OPS is inoperable; or 3.) conducting OPS testing periodically while at power.

Therefore, Con Edison has concluded that the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia (Acting).

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: May 31, 2001.

Description of amendment request: The proposed amendment would increase the allowed outage time (AOT) for one inoperable emergency diesel generator (EDG) from 72 hours to 14 days to allow the performance of various maintenance and repair activities while the plant is operating.

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

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

The proposed Technical Specification change to increase the EDG AOT from 72 hours to 14 days will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The EDGs are not accident initiators, and extending the EDG AOT will not impact the frequency of any previously evaluated accidents. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report (FSAR). In addition, extending the EDG AOT will not impact the consequences of an accident previously evaluated. The consequences of previously evaluated accidents will remain the same during the proposed 14 day AOT as during the current 72 hour AOT. The ability of the remaining EDG to mitigate the consequences of an accident will not be affected since no additional failures are postulated while equipment is inoperable within the Technical Specification AOT. The remaining EDG is sufficient to mitigate the consequences of any design basis accident. Therefore, the proposed change will not increase the probability or consequences of an accident previously evaluated.

The proposed Technical Specification change to allow verification of offsite circuit(s) within 1 hour prior to or after entering the condition of either an inoperable offsite source or inoperable EDG will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. Performing a verification of the offsite circuits does not require any equipment manipulations or operator actions that could cause a previously evaluated accident to occur. Providing the flexibility to verify offsite circuit availability before removing equipment from service will reduce the potential to establish an adverse plant configuration. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 FSAR. In addition, allowing an early verification of offsite circuit(s) will not impact the consequences of an accident previously evaluated. The consequences of previously evaluated accidents will remain the same whether the verification is performed immediately after, or just before, an EDG or offsite circuit is removed from service. The ability of the remaining power sources to mitigate the consequences of an accident will not be affected since no additional failures are postulated while equipment is inoperable within the Technical Specification AOT. The remaining power sources are sufficient to mitigate the consequences of any design basis accident. Therefore, the proposed change will not increase the probability or consequences of an accident previously evaluated.

The proposed Technical Specification changes associated with the requirements for the pressurizer heaters to be supplied by emergency power will not result in any change in plant design. These components will continue to be powered from Class 1E power sources. As a result, the operation and reliability of the pressurizer heaters will not be affected by the proposed changes. In addition, operation of the pressurizer heaters is not assumed to mitigate any design basis accident. The proposed changes will not cause an accident to occur and will not result in a change in the operation of any accident mitigation equipment. The design basis accidents remain the same postulated events described in the Millstone Unit No. 2 FSAR. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The additional change to add the requirement to verify that the steam driven auxiliary feedwater (SDAFW) pump is operable when one EDG is inoperable will ensure sufficient auxiliary feedwater capability is available if a loss of offsite power were to occur. Operation of the SDAFW pump will not be affected by the proposed change, and the SDAFW pump is not an accident initiator. Verifying operability of the SDAFW pump will not impact the frequency of any previously evaluated accidents. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 FSAR. The ability of the SDAFW pump to mitigate the consequences of an accident will not be affected. Therefore, the proposed change will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes to renumber action requirements and remove a footnote that is no longer valid will not result in any technical changes to the current requirements. Therefore, these additional proposed change[s] will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes to the Technical Specifications do not impact any system or component in a manner that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operators following an accident will not be significantly different. In addition, the proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed Technical Specification change to increase the EDG AOT from 72 hours to 14 days and allow verification of offsite circuit(s) within 1 hour prior to or after entering the condition of an inoperable offsite source or inoperable EDG does not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed Technical Specification change, in conjunction with the administrative controls, provides adequate assurance of the capability to supply power to the safety related Class 1E electrical loads thereby ensuring the accident mitigation functions will be maintained. The availability of offsite power combined with the availability of the Millstone Unit No. 3 Station Blackout diesel generator and the use of the Configuration Risk Management Program required by 10 CFR 50.65(a)(4) provide adequate compensation for the small incremental increase in plant risk of the proposed EDG AOT extension. This small increase in plant risk while operating is offset by a reduction in shutdown risk resulting from the increased availability and reliability of the EDGs during refueling outages, and avoiding transition risk incurred during unplanned plant shutdowns. In addition, the calculated risk measures associated with the proposed AOT are below the acceptance criteria defined in Regulatory Guide 1.177, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' dated August 1998. Therefore, the proposed change will not result in a significant reduction in a margin of safety.

The proposed Technical Specification changes associated with the requirements for the pressurizer heaters to be supplied by emergency power do not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The emergency power requirement for the pressurizer heaters, which came from the Three Mile Island (TMI) action item requirement item ll.E.3.1, Emergency Power Requirements for Pressurizer Heaters,'' of NUREG–0737, ''A Clarification of TMI Action Plan Requirements,'' will continue to be met. The pressurizer heaters are permanently connected to Class 1E power supplies as described in the Millstone Unit No. 2 FSAR. Therefore, these changes will not result in a significant reduction in a margin of safety.

The additional more restrictive change to add the requirement to verify that the SDAFW pump is operable when one EDG is inoperable will not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. Operation of the SDAFW pump will not be affected by the proposed change. Therefore, this change will not result in a significant reduction in a margin of safety.

The additional proposed changes to renumber action requirements and remove a footnote that is no longer valid will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in 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.

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385 Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

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: July 2, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated July 2, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: July 2, 2001.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated July 2, 2001.

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

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: July 9, 2001.

Description of amendment request: This Technical Specification (TS) change removes TS requirements that will no longer be applicable following replacement of the part-length control element assemblies (PLCEAs) with five-element full-length control element assemblies (CEAs) and removal of the four-element CEAs on the core periphery.

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

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

The proposed changes are administrative in nature and maintain the conservative restrictions on the operation of the CEAs. Chapter 15 of the Waterford 3 [Waterford Steam Electric Station, Unit 3] Safety Analysis Report identifies the analyses associated with the CEAs. These analyses are evaluated in the development of the Reload Analysis for each fuel cycle, and the appropriate limitations to insure acceptable analysis results are incorporated in the Core Operating Limits Report (COLR) for the fuel cycle. The modifications replacing the part-length CEAs with full-length CEAs and removing the four-element CEAs will be evaluated under the 10 CFR 50.59 process prior to implementation. The Reload Analysis and changes to the COLR are also evaluated under the 10 CFR 50.59 process prior to incorporating the identified changes.

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

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

The proposed changes introduce no new mode of plant operation and are considered to be administrative in nature. Operating experience has shown that the full-length CEAs are capable of controlling the axial power distribution function intended for the part-length CEAs. The part-length CEAs will be replaced with the same type of full-length CEAs used in the shutdown and regulating CEA groups. Removal of the four-element CEAs provides no mechanism for creating a new or different kind of accident.

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

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

The proposed changes may improve overall safety margins. Replacements of the part-length CEAs with full-length CEAs will allow Entergy [Operations, Inc.,] to credit these CEAs in the shutdown margins calculations. The worth of the four-element CEAs being removed from the core is relatively small in the modern low-leakage core design used at Waterford 3. Therefore, the removal of the four-element CEAs in combination with the replacement of the part-length CEAs with full-length CEAs will result in an overall increase in the net CEA worth. This will result in an increase in the available shutdown margin during reactor operation.

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 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: N. S. Reynolds, Esquire, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 26, 2001.

Description of amendment request: The proposed amendments would extend the dates specified in Operating License Sections 2.C(8) and 3.P, ''Pressure-Temperature Limit Curves,'' for Dresden Nuclear Power Station, Units 2 and 3, respectively.

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

The proposed change to the operating license to extend the limitations on the use of the [Pressure-Temperature] P–T limits does not affect the operations or configuration of any plant equipment. Thus, no new accident initiators are created by this change.

The proposed change extends the use of the pressure-temperature (P–T) limits for an additional cycle of operation on each unit. The P–T limits are based on the projected reactor pressure vessel (RPV) neutron fluence at 32 effective full power years (EFPYs) of operation. At the end of the next cycle of operation, the Dresden Nuclear Power Station (DNPS) units will have attained a maximum of 67.5% of the 32 EFPY operating times. Separately, we submitted a license amendment request to permit operation with an extended power uprate (EPU). Even with an approximately 17% increase in reactor power for one cycle due to the EPU, this provides significant margin to ensure that the current 32 EFPY fluence projection of 5.1×10^{17} n/cm^2 will not be exceeded. This ensures that the basis for proposed applicability of the P–T limits is conservative and that the RPV integrity is protected under all operating conditions. Therefore, neither the probability nor the consequences of an accident are increased.

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

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

The proposed change to the operating license to extend the limitations on the use of the P–T limits does not affect the operation or configuration of any plant equipment. The current P–T limits will remain valid and conservative during the proposed extension. Thus, no new or different accidents are created by this proposed change.

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

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

The proposed change extends the use of the P–T limits for an additional cycle of operation on each unit. The P–T limits are based on the projected RPV neutron fluence at 32 EFPYs of operation. At the end of the next cycle of operation, the DNPS units will have attained a maximum of 67.5% of the 32 EFPY operating times. In a separate license amendment request, ComEd submitted a license amendment request to permit operation with an extended power uprate (EPU). Even with an approximately 17% increase in reactor power for one cycle due to the EPU, this provides sufficient margin to ensure that the current 32 EFPY fluence projection of 5.1×10^{17} n/cm^2 will not be exceeded. This ensures that the basis for the P–T limits is conservative and therefore ensures that the reactor pressure vessel integrity is protected under all operating conditions. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: June 15, 2001 (RS–01–117).

Description of amendment request: The proposed amendments would allow the use of ATRIUM 10 fuel from Framatome Advanced Nuclear Fuel, Inc.

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

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

The proposed changes to LaSalle County Station, Unit 1 and Unit 2 Technical specification (TS), add the fuel analytical methods to TS Section 5.6.5, ''Core Operating Limits Report (COLR),'' that support insertion of Framatome Advanced Nuclear Fuel, Inc. (i.e., Framatome) ATRIUM 10 fuel.

LaSalle County Station Unit 1, is scheduled to load ATRIUM 10 fuel during its upcoming outage in November 2001. The proposed changes to TS Section 5.6.5 will add the fuel analytical methods that support the initial insertion of ATRIUM 10 fuel tot he list of methods used to determine the core operating limits. The addition of approved methods to TS Section 5.6.5 has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The NRC approved methods have been reviewed to ensure that the output accurately models predicted core behavior, have no affect on the type or amount of radiation released, and have no affect on predicted offsite doses in the event of an accident.

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

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

The proposed changes to TS Section 5.6.5 do not affect the performance of any LaSalle County Station structure, system, or component credited with mitigating any accident previously evaluated. The insertion of a new generation of fuel which has been analyzed with NRC approved methodologies will not affect the control parameters governing unit operation or the response plant equipment to transient conditions. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

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

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

The proposed changes to TS Section 5.6.5 will add the ATRIUM 10 fuel analytical methods to the list of methods used to determine the core operating limits. The additional methods have been previously approved by the NRC for use by licensees. The proposed changes do modify the safety limits or setpoints at which protective actions are initiated, and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

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

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

Attorney for licensee: Mr. Robert Helfrich, Senior Counsel, Nuclear, Mid-West regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515.

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: June 15, 2001 (RS–01–118).

Description of amendment request: The proposed amendments would modify Technical Specification (TS) Section 3.5.1, ''ECCS–Operating,'' Surveillance Requirement (SR) 3.5.1.8. The proposed changes will eliminate the requirement that the Automatic Depressurization System (ADS) designated Safety/Relief Valves (S/Vs) open during the manual actuation of the ADS and changes the SR frequency to require the testing of all required ADS manual actuation solenoids during the performance of SR 3.5.1.8 in lieu of on a staggered basis.

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:

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

The proposed changes modify Technical Specifications (TS) Section 3.5.1, ''ECCS—Operating,'' Surveillance Requirement (SR) 3.5.1.8. The proposed changes will eliminate the TS requirement that the Automatic Depressurization System (ADS) designated Safety/Relief Valves (S/RVs) open during the manual actuation of the ADS and rewords the SR frequency to require the testing of all required ADS manual actuation solenoids during the performance of SR 3.5.1.8 in place of testing on a staggered basis. The performance of ADS valve testing is not a precursor to any accident previously evaluated and does not change the manner in which the ADS is operated. Thus, the proposed changes to the performance of SR 3.5.1.8 do not have any affect on the probability of an accident previously evaluated.

The testing provides assurance that the ADS will functions as designed when actuated to depressurize the Primary Coolant System (PCS). The proposed changes to the surveillance requirement provide the same level of assurance regarding ADS reliability as the previous surveillance requirements. Accordingly, the consequences of an accident previously evaluated where the ADS was credited with mitigation is unchanged.

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

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

The proposed changes to SR 3.5.1.8 do not affect the performance of any LaSalle County Station structure, system, or component credited with mitigating any accident previously evaluated since the proposed changes will provide the same level of confidence concerning the functioning of the ADS as the current requirements. Furthermore, the proposed changes do not install any new equipment, introduce any new modes of system operation or failure mechanisms.

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

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

The proposed changes to SR 3.5.1.8 will allow the uncoupling of the ADS valve lever from the other components associated with the manual actuation of the ADS valve. The proposed changes will allow the testing of the manual actuation electrical circuitry, manual actuation solenoid and air control valve, and the actuator without causing the ADS valve to open. The ADS valves will continue to be manually actuated by the bench-test valve control system of the setpoint testing program. The proposed changes do not affect the valve setpoint or the operational criteria that directs the ADS valves to be manually opened during plant transients.

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

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

Attorney for licensee: Mr. Robert Helfrich, Senior Counsel, Nuclear, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515.

NRC Section Chief: Anthony J. Mendiola.

Exelon Energy Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: April 23, 2001

Description of amendment request: The proposed change would delete the loose parts monitoring system (LPMS) and the associated Technical Specifications (TSs) and Bases currently contained in the Limerick Generating Station, Units 1 and 2 Technical Specifications. The licensee bases its proposal to delete the LPMS on the conclusions of the Boiling Water Reactor Owners' Group Tropical Report NEDC–32975P, ''Regulatory Relaxation for BWR Loose Parts Monitoring Systems''.

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

This Technical Specification (TS) Change Request will delete the Loose Parts Monitoring System and the associated Technical Specifications and Bases currently contained in the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications. The Loose Parts Monitoring System (LPMS) is not an accident initiating system. The LPMS was designed in conformance with Regulatory Guide 1.133 (''Loose-Parts Detection Program for the Primary System of Light-Water-Cooled Reactors,'' Revision 1, May 1981), to detect and alarm for loose parts in the reactor coolant system. A secondary function of the system is to assist the operators in locating the detected loose parts. The LPMS is used for information purposes only and is not a safety-related system. The operators do not rely solely on this system or information provided by this system for the performance of any safety-related action. Review of the Updated Final Safety Analysis (UFSAR) indicates that this system is not relied upon by other systems for input or data. This is a monitoring system that does not perform any automatic or control functions, and is not relied upon for any accident or transient evaluation. The removal of the LPMS from operation will not increase the need for operator intervention or increase operator burden to support any system used to mitigate an accident under normal or off normal conditions. Therefore, the proposed changes will not significantly increase the probability of an accident previously evaluated.

The removal of the LPMS will not change or degrade the physical barriers or systems designed to contain radiation, and will have no affect on the on-site or off-site radiological conditions. Therefore, the proposed TS changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

This TS Change Request will delete the Loose Parts Monitoring System and the associated Technical Specifications and Bases currently contained in the LGS, Units 1 and 2, Technical Specifications. Removal of this system will not create a new mode of operation of the plant. The LPMS is a nonsafety-related monitoring system. The proposed changes do not create a system-level failure mode different than those that already exist. In addition, there are no operation or failure modes of the LPMS that are accident initiators. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This TS Change Request does not affect any safety limits or analytical limits. Also there are no changes to accident or transient core thermal hydraulic conditions, or fuel or reactor coolant boundary design limits, as a result of these proposed changes. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania.

Date of amendment request: June 1, 2001.

Description of amendment request: The proposed amendment would revise the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications (TS) 3.6.1.7 drywell average air temperature limit from 135 °F to 145 °F.

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

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

The increase in the allowable drywell average air temperature does not make any physical changes to the plant; it only permits the plant to operate at a higher drywell average air temperature, and therefore, does not increase the probability of an accident previously evaluated.

The LGS Mark II containment design was evaluated during Power Rerate using an initial temperature of 150 degrees Fahrenheit for the Loss-of-Coolant Accident (LOCA) due to an instantaneous double-ended rupture of a recirculation suction line. The results of this evaluation showed that the drywell air temperature does not exceed the limit of 340 degrees Fahrenheit post-accident and that the peak drywell pressure does not exceed the design limit of 55 psig. In addition, the containment analysis performed for Power Rerate also bounds the small break LOCA.

Since the proposed change allows a drywell air temperature that remains within the design analysis value, this proposed change does not increase the consequences of an accident previously evaluated in the Safety Analysis Report (SAR). This proposed change does not adversely affect mitigating systems, structures or components (SSC), and does not adversely affect the initial conditions of any accidents. Redundancy and diversity of mitigating systems are unchanged as a result of this proposed change. This proposed change does not affect onsite or offsite radiological consequences of any accident previously evaluated in the SAR.

Based on the above discussion, this proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The increase in the drywell average air temperature proposed by this TS change does not change any SSC of the plant. This TS change does not create new operating or failure modes.

Based on the above discussion, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change will allow the plant to operate at a higher drywell average temperature during normal operation. This change does not create additional heat loads or change the way any of the equipment is operated. The equipment will remain within the limitations of the equipment qualification (EQ) program, which is qualified/maintained based on operation at an average annual temperature of 145 degrees Fahrenheit.

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

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

Attorney for licensee: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Beaver County, Pennsylvania

Date of amendment request: March 28, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements to credit the soluble boron in the fuel storage pool analyses. This amendment would revise the index, modify TS 3.9.14, ''Fuel Storage—Spent Fuel Storage Pool,'' add TS 3.9.15, ''Fuel Storage Pool Boron Concentration,'' modify applicable Bases and revise Design Feature Section 5.3.1.1, ''Criticality.'' TS 3.9.14 would be modified by separating this specification into two specifications to support crediting soluble boron in the fuel storage pool. The revised TS 3.9.14 would provide controls for fuel assembly enrichment and burnup in the spent fuel pool and also include an increase in the maximum enrichment from 4.85 weight percent (w/o) to 5.0 w/o. A new TS 3.9.15 would provide control for soluble boron requirements in the spent fuel pool. Separating this specification into two specifications follows the general guidance provided in the improved standard TS (ISTS) of NUREG–1431.

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?

Because of the Boraflex deterioration that has been observed, the spent fuel racks have been reanalyzed neglecting the presence of Boraflex to allow storage of Westinghouse 17 x 17 fuel assemblies with nominal enrichments up to 5.0 weight percent (w/o) using credit for checkerboarding, burnup and soluble boron. The proposed changes will not have a significant impact on the safety of the plant or on the spent fuel storage pool and are consistent with the NRC approved changes identified for other plants (i.e., Prairie Island Units 1 and 2, Vogtle Units 1 and 2). Criteria set forth in Table 3.9–1 provide qualification requirements for fuel assembly storage to ensure the NRC acceptance criteria and accident analysis assumptions are satisfied. Increasing the enrichment from 4.85 w/o up to and including 5.0 w/o U–235 [uranium 235] has minor effects on the radiological source terms and subsequently the potential releases, both normal and accidental, are not significantly affected.

The proposed Technical Specification changes credit the use of soluble boron in the spent fuel pool criticality analyses. These criticality analyses were performed using the NRC approved methodology developed by the Westinghouse Owners Group (WOG) and described in WCAP–14416–NP–A, Revision 1, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' November 1996. The analysis includes evaluations that factor in the axial burnup bias correction and utilizing identified conservatisms in the analysis demonstrate that K_{eff} remains less than or equal to the design limits.

The proposed changes do not involve a change to plant equipment and do not affect the performance of plant equipment used to mitigate an accident. They do not affect the operation of the spent fuel pool cooling system or any other system and are consistent with applicable analyses including [those associated with postulated] fuel handling accidents. They will not affect the ability of any system to perform its design function; therefore, the proposed 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?

There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. No new accident scenarios, transient precursors, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not introduce any adverse effects or challenges to any safety-related systems.

The potential criticality accidents have been reanalyzed to demonstrate that the pool remains subcritical. Soluble boron has been maintained in the fuel storage pool water since its initial operation. The possibility of a fuel storage pool dilution is not affected by the proposed changes to the Technical Specifications. Therefore, implementation of Technical Specification controls for the soluble boron will not create the possibility of a new or different kind of accidental pool dilution.

With credit for soluble boron now a major factor in controlling subcriticality, an evaluation of fuel storage pool dilution events was completed. This evaluation concluded that no credible events would result in a reduction of the criticality margin below the 5% margin recommended by the NRC. In addition, the No Soluble Boron 95/95 probability/confidence level criticality analysis assures that dilution to 0 ppm [parts per million] will not result in criticality.

The proposed Technical Specification changes ensure the maintenance of the fuel pool boron concentration and storage configuration. Therefore, the proposed changes will not create the possibility of any 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 changes do not affect the acceptance criteria for any analyzed event nor impact any plant safety analyses since the analysis assumptions are not changed. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since the proposed changes do not affect equipment required to mitigate design basis accidents described in the Updated Final Safety Analysis Report. The Technical Specifications continue to assure that applicable operating parameters are maintained within required limits.

The proposed changes to the fuel storage pool boron concentration and storage requirements will provide adequate margin to assure that the fuel storage array will always remain subcritical by the 5% margin recommended by the NRC. These limits are based on a criticality analysis performed in accordance with NRC approved Westinghouse fuel storage rack criticality analysis methodology.

While criticality analysis utilized credit for soluble boron, the storage configurations have been defined using K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to offset off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the fuel storage pool K_{eff} is maintained less than or equal to 0.95.

The spent fuel pool boron dilution analysis concludes that an unplanned or inadvertent event which would result in dilution of the spent fuel pool boron concentration from 2000 ppm to 450 ppm is not a credible event. This conclusion is based on the substantial volume of unborated water required to dilute the pool and the fact that a large dilution event would be readily detected by plant personnel via alarms, flooding in the fuel handling building or detected during normal operator rounds through the spent fuel pool area.

The margin of safety depends upon maintenance of specific operating parameters within design limits. The Technical Specifications continue to require that these limits be maintained and provide appropriate remedial actions if a limit is exceeded. The maintenance of these limits continues to be assured through performance of surveillances. Therefore, the plant will be maintained within the analyzed limits and the proposed changes will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Richard P. Correia, Acting.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station (DBNPS), Unit 1, Ottawa County, Ohio

Date of amendment request: May 22, 2001.

Description of amendment request: The proposed amendment would revise the once-through steam generator (OTSG) tube repair roll requirements to (1) Utilize updated limiting tensile tube loads, (2) define new exclusion zones within the steam generator in which the application of the repair roll is prohibited, (3) allow the repair roll to be used in the lower tubesheet area, (4) remove the limitation of only one repair roll per OTSG tube, and (5) replace the requirement that the repair roll be one inch in length with a requirement that the repair roll be installed in accordance with Framatome Technologies Incorporated Report BAW–2303P, revision 4, ''OTSG Repair Roll Qualification Report.''

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because testing and analysis have shown the once-through steam generator (OTSG) tube repair roll process under the proposed revised Technical specification (TS) Surveillance Requirement (SR) 4.4.5.4 ensures the new pressure boundary joint created by the repair roll process provides adequate structural and leakage integrity for all normal operating and accident conditions. In addition, the removal of the name ''Babcock & Wilcox'' is an administrative change to reflect that Framatome ANP has succeeded the Babcock & Wilcox Company. Therefore, the proposed changes to SR 4.4.5.4 will not increase the probability of a previously evaluated accident.

The proposed change to TS Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase in the probability of an accident previously evaluated.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the repair roll process under the proposed revised SR 4.4.5.4 ensures the new pressure boundary joint created by the repair roll process provides adequate structural and leakage integrity under all accident conditions. Any leakage resulting from repair roll joint slippage under accident conditions will be accounted for to ensure that the post-accident OTSG leakage will not exceed that assumed in the accident analyses. Should a repaired tube fail, the radiological consequences would be bounded by the existing Steam Generator Tube Rupture analysis.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not involve an increase to the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because there will be no change in the operation of the steam generators or connecting systems as a result of the repair roll process added by the proposed changes to SR 4.4.5.4. The physical changes in the steam generators associated with the repair roll process have been evaluated and do not create the possibility for a new or different kind of accident from any accident previously evaluated, i.e., the physical change in the steam generators is limited to the location and accident slip behavior of the primary to secondary boundary within the tubesheet. Accordingly, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not create the possibility of any new or different kind of accident.

3. Not involve a significant reduction in a margin of safety because tubes with primary system to secondary system boundary joints created by the repair roll process have been shown by testing and analysis to satisfy all structural, leakage, and heat transfer requirements. The additional testing of tubes repaired by the repair roll process under existing SR 4.4.5.9 provides continuing inservice monitoring of these tubes such that inservice degradation of tubes repaired by the repair roll process will be detected. Therefore, the changes to SR 4.4.5.4 to modify the repair process do not reduce and margin of safety.

The proposed change to Bases 3/4.4.5 reflects the changes proposed to its associated SR, and does not reduce the margin of safety.

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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.

Attorney for licensee: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: July 18, 2001.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.9.4, Containment Penetrations. TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. The proposed changes to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: (a) The equipment door is capable of being closed with four bolts, (b) the plant is in MODE 6 with at least 23 feet of water above the reactor vessel flange, and (c) a designated crew is available to close the door. The basis for the proposed changes is a reanalysis of the limiting design basis Fuel Handling Accident, using an Alternate Source Term in accordance with 10 CFR 50.67 and Regulatory Guide (RG) 1.183.

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

The proposed changes to TS 3.9.4 would allow the containment equipment door and both doors of each containment airlock to be open during fuel movement or core alterations. Currently, the equipment door is closed with four (4) bolts during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. The containment equipment door is not an initiator of an accident. Whether the containment equipment door is open or closed during fuel movement and core alterations has no effect on the probability of any accident previously evaluated.

Allowing the containment equipment door to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 50.67 and RG 1.183. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the containment equipment door after the containment is evacuated, which would occur in much less than the two hours assumed in the analysis.

The changes being proposed do not affect assumptions contained in other plant safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to Technical Specification 3.9.4, ''Containment Building Penetrations,'' affect a previously evaluated fuel handling accident. Both the current and the revised fuel handling accident analyses assume that all of the iodine and noble gases that become airborne, escape and reach the site boundary and low population zone with no credit taken for filtration, for the containment building barrier, or for decay or deposition. Since the proposed changes do not involve the addition or modification of equipment nor alter the design of plant systems, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The margin of safety has not been significantly reduced. The calculated dose is well within the limits given in 10 CFR Part 50.67 and RG 1.183. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Patrick M. Madden.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: July 18, 2001.

Description of amendment request: The proposed amendments would change the title of the corporate executive responsible for overall plant nuclear safety from ''President-Nuclear Division'' to ''Chief Nuclear Officer,'' in Technical Specification (TS) Section 6.0.

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

The proposed amendments are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety, and would not involve a significant increase in the probability or consequences of an accident previously evaluated. These amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect TS that preserve safety analysis assumptions. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the TS are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety in the Turkey Point Units 3 and 4 TS, and would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the administrative changes since the proposed changes do not involve the addition or modification of equipment, nor do they alter the design or operation of affected plant systems, structures, or components.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature, changing the title of the corporate executive responsible for overall plant nuclear safety in the Turkey Point Units 3 and 4 TS, and would not reduce any of the margins of safety. The operating limits and functional capabilities of the affected systems, structures, and components remain unchanged by the proposed amendments.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420

NRC Section Chief: Patrick M. Madden.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: April 11, 2001.

Description of amendment request: The proposed action would modify Technical Specification (TS) sections 4.2, ''Fuel Storage,'' and 5.6.5, ''Spent Fuel Pool Water Chemistry Program,'' by adding applicability statements that these sections apply only when irradiated fuel is stored in the fuel storage pool. The applicability statements will allow timely dismantlement of the fuel storage pool following removal of the last irradiated fuel assembly from the fuel storage pool to the onsite independent spent fuel storage installation.

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

The proposed change does not:

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

The requested license amendment involves addition of applicability statements to the design features for fuel storage and the program requirements for the spent fuel pool water chemistry program. These applicability statements will make the respective design features and program requirements applicable whenever irradiated fuel is stored in the fuel storage pool. Once irradiated fuel has been completely removed from the fuel storage pool and transferred to certified dry storage containers under a general [10 CFR] Part 72 license, these design features and program requirements for the fuel storage pool are no longer necessary. These design features include: the maximum allowable Uranium-235 enrichment in fuel assemblies stored in racks; minimum acceptable margin to criticality allowed in the design of the spent fuel racks; the nominal fuel cell spacing in the spent fuel rack design; the minimum allowable drainage prevention design elevation; the fuel assembly loading capacity of the fuel storage pool and the specified storage locations within the fuel storage pool for different fuel enrichments and burnup periods; and the maximum allowable number of standard fuel assemblies in consolidated form. The program requirements consist of the establishment, implementation and maintenance of a water chemistry program for the fuel storage pool to minimize the potential effects of corrosion.

The corresponding design features and program requirements for fuel storage in dry storage containers are specified in the container's certificate of conformance and safety analysis report. The corresponding design features currently include: fuel loading positions; fuel assembly limits including consolidated fuel and minimum cooling times versus burnup/initial enrichments. Descriptions of other design features of the UMS [Universal Multipurpose System] Storage System are found in the NAC [Nuclear Assurance Corporation]—UMS ® SAR [Safety Analysis Report]. The corresponding program requirements currently include specifications for canister vacuum drying pressure and helium backfill pressure which ensure that a sufficiently inert environment is produced within the canister to preclude or inhibit corrosion.

Since the design features and program requirements associated with fuel storage in the fuel storage pool do not significantly contribute to accident prevention or mitigation following the complete removal of irradiated fuel and since the corresponding design features and program requirements for fuel storage in dry storage containers are specified and controlled under other applicable license documents, these changes do not significantly increase the probability or the consequences of an accident previously evaluated.

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

The requested amendment involves the addition of applicability statements which will have the effect of making certain design features and program requirement associated with the fuel storage pool inapplicable when the fuel storage pool is no longer used for fuel storage. The corresponding design features and program requirements for fuel storage in dry storage containers are adequately specified in applicable license documents. The elimination of these design features and program requirements following complete removal of irradiated fuel from the fuel storage pool does not result in any new or different accident initiators from those already assumed in accidents previously evaluated, nor does it exacerbate any such accidents. Therefore, these changes do 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.

The safety margins produced as a result of the specification of design features and program requirements for fuel storage in the fuel storage pool are adequately maintained in corresponding design features and program requirements associated with fuel storage in dry storage containers. These corresponding design features and program requirements are specified in the dry storage container's certificate of conformance and safety analysis report. Therefore, these changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Joseph Fay, Esquire, Maine Yankee Atomic Power Company, 321 Old Ferry Road, Wiscasset, Maine 04578.

NRC Section Chief: Robert A. Gramm.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: April 6, 2001.

Description of amendment request: The proposed amendment would revise technical specification (TS) 5.5.10, ''Technical Specifications Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (64 FR 53582) dated October 4, 1999. TS 5.5.10.b.2. would be revised to state: ''A change to the updated final safety analysis report or Bases that requires Nuclear Regulatory Commission approval pursuant to 10 CFR 50.59.'' In TS 5.5.10.b, a minor editorial change replaces the phrase ''changes do not involve'' with ''changes do not require.'' This change is consistent with the Nuclear Energy Institute Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–364 Revision 0, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59.''

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

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

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revision of 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph 10 CFR 50.59 (c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature based on the revision to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: September 26, 2000, as supplemented on October 6, 2000, and May 21, 2001. This notice supersedes a previous notice (65 FR 69065) published on November 15, 2000, that was based on the licensee's application for amendment dated September 26, 2000, as supplemented on October 6, 2000.

Description of amendment request: The proposed change would amend the Salem Nuclear Generating Station (Salem) Unit Nos. 1 and 2 Technical Specifications (TSs) to increase the as-found setpoint tolerance for the Pressurizer Safety Valves (PSV) from ±1% to ±3%; increase the as-found setpoint tolerance for the Main Steam Safety Valves (MSSV) from ±1% to ±3%; change the required action for reducing power when one or two MSSVs are inoperable; change the required action for three inoperable MSSVs to include a requirement to decrease the Power Range Neutron Flux High trip setpoint in addition to reducing power; and remove specifications and references related to plant operation with three Reactor Coolant System loops. The associated TS Bases sections will also be amended to reflect the TS changes.

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

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

Changing the pressurizer and main steam safety relief valve lift setpoint tolerance from ±1% to ±3% does not significantly increase the probability of any accident previously evaluated. The only events initiated by the opening of these safety valves are the accidental depressurization of the Reactor Coolant System and accidental depressurization of the Main Steam System. These events are a result of an inadvertent lifting of these valves and do not depend on the safety valve lift setpoint or tolerance. Therefore, the likelihood that either of these events will occur has not been increased.

Analyses associated with the limiting overpressurization transients (Loss of External Electrical Load and/or Turbine Trip, and Single Reactor Coolant Pump Locked Rotor) have been performed that demonstrate that increasing the Pressurizer Safety Valve and Main Steam Safety valve lift setpoint tolerance to ±3% would result in primary and secondary side pressure responses less than the acceptance criteria of 110% of the design pressure. Therefore, since the proposed setpoint tolerance increase would not adversely impact current accident analysis assumptions, the proposed change would not result in an increase in consequences of an accident previously evaluated.

For operation with one or two inoperable main steam safety valves in one or more steam generators, changing the required action from a reduction of the power range high neutron flux trip setpoint to a reduction of the allowable reactor power level will not increase the consequences of any accident. With one or two inoperable Main Steam Safety Valves, the Loss of External Electrical Load and/or Turbine Trip event becomes limiting in terms of secondary side pressurization. The high flux trip does not provide any mitigation for this event. Other events limiting at power, that require the power range trip for mitigation, assume a safety analysis trip setpoint of 118% (based on a nominal trip setpoint of 109%) regardless of the initial power level. Therefore, the proposed change does not impact any of the accident analysis assumptions.

During an RCCA (Reactor Cluster Control Assembly) Bank Withdrawal at Power event, the Main Steam Safety Valves may lift to ensure secondary side pressure remains below the allowable limit. This is especially true for events initiated from partial power conditions and slow reactivity insertion rates, where the reactor trip is from Overtemperature ΔT (OTDT). Protection for this event is provided by a reactor trip on OTDT, not by the power range—high neutron flux trip. Thus, the proposed change does not affect the mitigative actions for this accident. Therefore, the consequences of an RCCA are unaffected.

For three inoperable main steam safety valves in one or more steam generators, the addition of a requirement for a lower Power Range Neutron Flux High trip setpoint ensures the proposed change does not increase the consequences of this postulated accident.

The current Salem licensing basis for the Spurious Activation of the Safety Injection System credits operator action to unblock a pressurizer Power Operated Relief Valve prior to the water solid pressurizer reaching the safety valve lift setpoint. The analyses that determined the time at which the safety valve would reach its pressure setpoint covered the −3% tolerance. Since this would conservatively result in the earliest opening time, there was no need to consider the positive side of the tolerance. The results of the analyses indicate that the allowable operator action time has not changed, such that water relief continues to occur through the Power Operated Relief Valves and not through the PSVs. As such the consequences of this event have not changed as a result of the proposed change.

Increasing the MSSV lift setting tolerance may result in increased secondary side backpressure for the Auxiliary Feedwater Pumps. However, analyses have demonstrated that with the elevated backpressures that could result from increasing the MSSV setpoint upper tolerance to +3%, the Auxiliary Feedwater Pumps would still provide greater than the minimum flow required to mitigate events in which normal feedwater is not available, a Loss of Normal Feedwater and a Loss of Offsite Power to Station Auxiliaries.

In terms of radiological consequences, the current design and licensing basis analyses that include steaming through the MSSV bound the proposed lift setpoint tolerance change.

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

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

The proposal will result in a change in the allowed Pressurizer Safety Valve and Main Steam Safety Valve lift setpoint tolerance range. No physical changes to these valves or to their nominal lift setpoint is required. These valves are assumed to malfunction only as the initiator for the accidental depressurization of the existing Reactor Coolant System or Main Steam System accident analyses. An increased lift setpoint tolerance range does not change the assumption of these depressurization events nor create a new type of event.

Requiring a reduction in reactor thermal power or a reduction in reactor thermal power in conjunction with a reduction in Power Range Neutron Flux High Trip setpoint in the event of inoperable MSSV is consistent with the existing analysis assumptions. Initiation of any Salem Updated Final Safety Analysis Report (UFSAR) analyzed event at a power level less than full power is bounded by those events analyzed at full power, or specifically analyzed at the limiting power level, and does not constitute a new or different kind of accident. Also, no changes are being made to the power range high flux trip setpoint that will make it inconsistent with any analytical assumption.

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

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

Analyses performed demonstrate that the proposed increase in the Pressurizer Safety Valve and MSSV lift pressure setpoint tolerance from ±1% to ±3% will provide primary and secondary side pressure responses to the anticipated operational occurrences and design basis accidents that are within the existing margins of safety. The limiting overpressurization transients, Loss of External Electrical Load and/or Turbine Trip, and Single Reactor Coolant Pump Locked Rotor, stay well within the acceptance criteria of 110% of the design pressure.

For operation with one or two inoperable MSSVs in one or more steam generators, the proposed reduction in reactor thermal power will ensure that current margins are maintained. The current requirement to reduce the power range high neutron flux trip setpoint does not affect the margin of safety since this trip does not provide any mitigation for the limiting secondary system pressurization event, Loss of External Electrical Load and/or Turbine Trip with one or two inoperable MSSVs.

Specific accident analyses for RCCA Bank Withdrawal at Power scenarios demonstrate that a reactor trip on OTDT, in conjunction with the available relief capacity that exists with up to two inoperable safety relief valves on each steam generator, results in a secondary side pressurization within existing margins.

For three inoperable MSSVs in one or more steam generators, thermal reactor power must be reduced in conjunction with a reduction in the Power Range Neutron Flux High trip setpoint to ensure pressurization of the main steam system remains within current analysis margins.

The current licensing basis for the Spurious Activation of the Safety Injection System credits operator action to unblock a pressurizer Power Operated Relief Valve prior to the water solid pressurizer reaching the Pressurizer Safety Valve lift setpoint. As the PSVs are not designed for water relief, failure to unblock a Power Operated Relief Valve before reaching the Pressurizer Safety Valve lift setpoint would result in water relief and likely failure of the Pressurizer Safety Valve to reseat. This condition would escalate the Spurious Activation of the Safety Injection System (Condition II event) into a small break Loss Of Coolant Accident (Condition III event). The analyses that determined the time at which primary system pressure would reach the Pressurizer Safety Valve setpoint bound the −3% tolerance. Since the Pressurizer Safety Valve would not fail due to water relief, there is no reduction in the margin of safety for this event.

Increasing the Main Steam Safety Valve lift setpoint tolerance may result in increased secondary side backpressure for the Auxiliary Feedwater System. However, analyses have demonstrated that under degraded Auxiliary Feedwater Pump performance, and with secondary side backpressure corresponding to 103% of the lowest MSSV setpoint, the Auxiliary Feedwater System can provide greater than the minimum flow required to mitigate those events where normal feedwater is not available, a Loss of Normal Feedwater and a Loss of Offsite Power to Station Auxiliaries.

Therefore the proposed changes to the Technical Specifications do not involve a significant reduction in a margin of safety.

Based on the NRC staff's 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.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

South Carolina Electric &Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: May 24, 2001.

Description of amendment request: The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirement (SR) 4.7.1.2 would be revised to include the emergency feedwater system automatic isolation valves into the SRs. SR 4.7.1.2.b would include verification of the functional capability of the check valves in the instrument air system supplying the six new automatic isolation valves. SR 4.7.1.2.c.2 would include the six new automatic isolation valves into the requirement that assures critical valves can be closed and held closed when normal instrument air is unavailable.

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 addresses necessary changes to the VCSNS Technical Specification[s] (TS) 4.7.1.2.b and 4.7.1.2.c.2 associated with the installation of six new automatic isolation valves in the EF [emergency feedwater] system. The TS [need] to be changed to assure the same level of operability for the EF system as exists with the present day configuration.

The only Final Safety Analysis Report (FSAR) analyzed accident for which the EF system could contribute as an initiator would be minor secondary line break, as described in Section 15.3.2. The addition of isolation valves in the EF piping to the steam generators will not increase the likelihood of a pipe break, since the addition will be in accordance with the same codes and standards as the corresponding, existing portions of the system. Piping stress analyses have demonstrated the addition of these valves does not result in the need to postulate any additional pipe breaks.

The accidents analyzed in the FSAR, which rely on EF to mitigate consequences, are loss of normal feedwater, loss of off-site power, and major secondary system pipe ruptures. The addition of these automatic isolation valves will eliminate the need for operator action to manually close a flow control valve in response to a major secondary system line break. The elimination of operator manual action is accomplished by the addition of a new pneumatically operated isolation valve in series with each of the six existing flow control valves. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change does not result in changes to actual operating pressures, flow rates, flow paths, or system interfaces. There are no alterations to system operability requirements. The existing system alarm setpoints are not affected, as is the information available to the operators. The addition of six new isolation valves will not change system design criteria and the surveillance testing will be the same as for the existing flow control valves.

This change does not introduce any new or different kind of failure mechanisms or limiting single failures. Piping analysis has concluded that no new pipe break locations or break sizes will result from this change. Equipment protection features are not impacted, the frequency of pump and valve operation remains the same. Independence and redundancy are actually improved. Therefore, this proposed change would not create the possibility of an accident of a different type.

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

The design basis for the EF system is to assure the required flow and pressure to remove decay heat from the core under the worst postulated conditions. An additional function of the system is to isolate flow to a faulted SG [steam generator] within the time assumed in the safety analysis. The proposed change eliminates the need for operators to take actions to manually close the flow control valves in the event of a single failure.

The proposed change will create a surveillance requirement for the new isolation valves that is the same as the existing flow control valves. The acceptance criteria will assure the operability of these valves. The design and installation of these isolation valves will maintain the requirements for independence, redundancy, separation and testability. The margins assumed in the safety analysis will be enhanced by this proposed change. Due to the automatic isolation capability, additional water will be available for the intact SGs and a reduced mass will be available to be released into the containment building. No credible single failure will be capable of preventing isolation of a faulted SG upon a high flow signal.

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.

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Uni No. 1, Fairfield County, South Carolina

Date of amendment request: June 19, 2001.

Description of amendment request: This request proposes to change Technical Specification (TS) Section 3.4.6.2, including its Bases, to increase the allowed operational leakage for Reactor Coolant System (RCS) Pressure Isolation Valves (PIV). The present criteria of 1 gallon per minute for all size valves would be changed to the industry standard of 0.5 gallons per minute per nominal inch of valve size, up to a maximum of 5 gallons per minute per valve, consistent with NUREG–1431. This request also proposes to revise Table 3.4–1 to reflect the allowable leakage rates for each PIV.

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

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

This proposed change provides a more appropriate Pressure Isolation Valve (PIV) allowable leakage criteria in consideration of the safety significance and design capabilities of the plant as determined by the improved standard technical specification industry effort.

The TS leakage limit for PIVs is 0.5 gallon per minute per nominal inch of valve size with a maximum limit of 5 gallons per minute. The previous criteria of 1 gallon per minute for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and can result in higher personnel radiation exposures due to unwarranted rework and retesting. An NRC sponsored study concluded a leakage rate limit based on the valve size was superior to a single allowable value.

The revision to a leakage criterion related to valve size is acceptable because associated systems that have larger valves also have greater pressure relief capability. The new criteria allows for leakage above 1 gallon per minute, although limited to a maximum of 5 gallons per minute, because the isolated low pressure system will not be overpressurized based on [its] relief capacity being greater than [its] allowed leakage limit. Therefore, the proposed change to the Limiting Condition for Operation will result in lower radiation exposures to personnel and a superior leak rate limit based on valve size as compared to a single allowable value.

Since this proposed revision would continue to support the required safety functions, without modification to the plant features, neither the probability nor the consequences of an accident are increased.

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

The proposed revision is not a result of changes to plant equipment, system design, testing methods, or operating practices. The modified LCO [limiting condition for operation] requirement will allow some relaxation of the current operability criteria for the PIVs, consistent with NUREG–1431. This change provides a more appropriate requirement in consideration of the safety significance and design capabilities of the plant as determined by the improved standard technical specification industry effort. Since the functions of the associated systems will continue to perform without change, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. This proposed change does not involve a significant reduction in a margin [of] safety.

The proposed revision to the PIV leakage acceptance criteria will not result in changes to system design or setpoints that are intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems. These systems will continue to operate without change and only the associated allowable leakage criteria has been altered.

Since the setpoints and design features that support the margin of safety are unchanged and actions for inoperable systems continue to provide appropriate time limits and compensatory measures, the proposed changes will not significantly reduce the margin of safety.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of amendment request: July 25, 2001 (TS–415).

Description of amendment request: The proposed amendment would delete Technical Specification Action Statement 3.3.1.1.I.2, which limits plant operation to 120 days in the event of the inoperability of the Oscillation Power Range Monitor (OPRM) trip system at Browns Ferry Nuclear Plant, Unit 3 (BFN). For this situation, the proposed change would allow plant operation to continue if the existing TS Required Action 3.3.1.1.I.1, to implement an alternate means to detect and suppress thermal hydraulic instability oscillations, were taken.

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:

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

The OPRM function is not considered as an initiator of any previously analyzed accident. Therefore, this proposed change does not significantly increase the probability of such accidents. This proposed change would allow the use of existing well-established alternate methods to detect and suppress thermal hydraulic instability oscillations. Considering that multiple Boiling Water Reactors plants, including BFN, have satisfactorily operated using alternate stability monitoring methods for extended periods of operation prior to the installation of OPRM systems, it is concluded these measures are adequate. Therefore, the consequences of a previously analyzed accident would not be significantly increased.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change would allow the use of existing alternate methods to detect and suppress thermal hydraulic instability oscillations to continue to operate the reactor in the event of the inoperability of the OPRM system. Considering that multiple Boiling Water Reactors plants, including BFN, have satisfactorily operated using alternate stability monitoring methods for extended periods of operation, it is concluded these measures are adequate, and that the proposed change does not significantly reduce the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Patrick M. Madden (Acting).

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to pdr@nrc.gov.

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: March 1, 2001, as supplemented on June 27, 2001.

Brief description of amendment: The proposed amendment revised the Technical Specifications (TSs) to change the frequency of closure time testing of the main steam isolation valves (MSIVs). These tests may now be conducted during each cold shutdown unless this test has been performed within the past 92 days.

Date of Issuance: July 17, 2001.

Effective date: 7/17/01 and shall be implemented within 30 days of issuance.

Amendment No.: 221.

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

Date of initial notice in **Federal Register**: April 18, 2001 (66 FR 19999).

The June 27, 2001, letter provided ''camera-ready'' TS pages and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 17, 2001.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Inc., Docket No. 50–003, Indian Point Nuclear Generating Station, Unit 1

Date of amendment request: October 5, 2000, as supplemented by letters dated June 27, 2001.

Brief description of amendment: The amendment revised Technical Specification Sections 3.2.1.a, 3.2.1.e, and 3.2.1.f to relocate administrative controls to the Quality Assurance Program Description.

Date of issuance: July 23, 2001.

Effective date: 30 days from the date of issuance.

Amendment No: 49.

Facility Operating License No. DPR–5: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 29, 2000 (65 FR 71134) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 8, 2001.

Brief description of amendment: The amendment revises the frequency of the Technical Specification (TS) surveillance requirement to check the movement of the control rods. Specifically, the frequency listed for this requirement in TS Table 4.1–3, ''Frequencies for Equipment Tests,'' is changed from ''every 31 days'' to ''quarterly during reactor critical operations.''

Date of issuance: July 18, 2001.

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

Amendment No.: 217.

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

Date of initial notice in **Federal Register**: June 12, 2001 (66 FR 31704).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 18, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Beaver County, Pennsylvania

Date of application for amendments: December 27, 2000, as supplemented on March 28, April 12, June 9, June 13, and June 29 (3), 2001. The addition of a Technical Specification (TS) Bases control program was requested on March 28, 2001.

Brief description of amendments: These amendments allow: (1) Revisions to reactor trip and engineered safety feature actuation setpoints and allowable values, (2) implementation of the revised thermal design procedure, (3) relocations of TS requirements to the core operating limits report, (4) relocation of TS requirements to the licensee requirements manual, (5) miscellaneous editorial changes. In addition, License Condition 2.(C).(3) regarding less than 3-loop operation was deleted.

Date of issuance: July 20, 2001.

Effective date: Immediately and to be implemented within 120 days.

Amendment Nos.: 239 and 120.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications and License.

Date of initial notice in **Federal Register:** April 18, 2001 (66 FR 20002) for the December 27, 2000, amendment request. A portion of a March 28, 2001, amendment request was also issued in this amendment. The date of the initial notice for the March 28, 2001, amendment request was June 20, 2001 (66 FR 33111).

The March 28, April 12, June 9, June 13, and June 29 (3), 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 20, 2001.

No significant hazards consideration comments received: No.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: January 4, 2001, as supplemented by letters dated March 12 and April 4, 2001.

Brief description of amendment: The amendment revises License Condition 2.B.(6)(d) to reference revisions to the Physical Security Plan, Guard Training and Qualification Plan, and Safeguards Contingency Plan.

Date of issuance: July 25, 2001.

Effective date: As of the date of issuance and shall be implemented 30 days from the date of issuance.

Amendment No.: 165.

Facility Operating License No. DPR–36: The amendment revised the License.

Date of initial notice in **Federal Register:** March 7, 2001 (66 FR 13805).

The March 12, 2001, supplemental letter superseded certain aspects of the January 4, 2001, amendment request, as described in the original **Federal Register** notice (FRN), but did not change the initial no significant hazards consideration determination (NSHCD). The April 4, 2001, supplemental letter provided clarifying information that did not change the scope of the original FRN or the initial NSHCD.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 5, 2001; as supplemented on April 19, 2001.

Brief description of amendment: The amendment revises Section 3.6.1.3, ''Primary Containment Isolation Valves,'' those portions regarding requirements for excess flow check valve surveillance testing.

Date of issuance: July 12, 2001.

Effective date: As of the date of issuance and shall be implemented prior to startup from Refueling Outage 8, currently scheduled for approximately spring 2002.

Amendment No.: 96.

Facility Operating License No. NPF–69: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15927).

The April 19, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: December 5, 2000, as supplemented June 28, 2001.

Brief description of amendment: The amendment implements programmatic controls for radiological effluent technical specifications (RETS) in the administrative section of the Technical Specifications (TSs) and relocates the procedural details of the RETS to the offsite dose calculation manual, the process control program, or other new programs, consistent with the guidance of Standard TSs (NUREG–1433) and Nuclear Regulatory Commission Generic Letter 89–01.

Date of issuance: July 24, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 120.

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

Date of initial notice in **Federal Register:** February 7, 2001 (66 FR 9385).

The June 28, 2001, supplement provided corrected TS pages to reflect the inclusion of amendments approved subsequent to the December 5, 2000, application, to correct a typographical error on one TS page, and to make a terminology change from ''site'' to ''unit'' on one TS page. The supplemental information did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: December 13, 2000, as supplemented July 3, 2001.

Brief description of amendment: The amendment revises Technical Specification 3.8/4.8 to clarify the air ejector offgas activity sample point and operability requirements.

Date of issuance: July 25, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 121.

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

Date of initial notice in **Federal Register:** January 24, 2001 (66 FR 7685).

The supplemental information did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 2001.

No significant hazards consideration comments received: No.

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 4, 2001.

Brief description of amendments: The amendments deletes Technical Specifications (TS) Section 5.5.3, ''Post Accident Sampling,'' for Diablo Canyon Nuclear Power Plant, Units 1 and 2, and thereby eliminate the requirements to have and maintain the post-accident sampling systems (PASS). The amendment for Unit 1 also deletes PASS-related License Condition 2.C(6).e from Facility Operating License DPR–80.

Date of issuance: July 13, 2001.

Effective date: July 13, 2001, to be implemented within 90 days from the date of issuance.

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

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

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31712).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 23, 2000, and supplemental letters dated January 11 and April 16, 2001.

Brief description of amendment: The amendment deletes the definitions of Site Boundary and Unrestricted Area from the technical specifications and makes related conforming changes.

Date of issuance: July 13, 2001.

Effective date: July 13, 2001, to be implemented within 30 days from the date of issuance.

Amendment No.: 128.

Facility Operating License No. DPR–54: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 7, 2001 (66 FR 13806).

The January 11 and April 16, 2001, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 13, 2001.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendments: May 14, 2001 (TS 01–02).

Brief description of amendments: This amendment revised License Condition 2.C.(9)(d) in Operating License DPR–77 for the Sequoyah Nuclear Plant. The revised license condition now references a licensee letter that specifies a minimum voltage threshold for steam generator tube eddy current inspections.

Date of issuance: July 18, 2001.

Effective date: July 18, 2001.

Amendment Nos.: 270.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the Operating License.

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29362).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 18, 2001.

No significant hazards consideration comments received: No.

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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to pdr@nrc.gov.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By September 7, 2001, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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 20555–001, and to the attorney for the licensee.

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

Tennessee Valley Authority, Docket Nos. 50–260, Browns Ferry Nuclear Plant, Units 2 Limestone County, Alabama

Date of amendment request: July 25, 2001.

Brief description of amendment: The proposed amendment deletes TS Required Action 3.3.1.1.I.2, which limits plant operation to 120 days in the event of the inoperability of the Oscillation Power Range Monitor trip system. For this situation, the proposed change would allow plant operation to continue if the existing TS Required Action 3.3.1.1.I.1, to implement an alternate means to detect and suppress thermal hydraulic instability oscillations, were taken.

Date of issuance: July 25, 2001.

Effective date: July 25, 2001.

Amendment No.: 273.

Facility Operating License No. NPF–90: Amendment revises the TS. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration, are contained in a Safety Evaluation dated July 25, 2001.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902.

NRC Section Chief: Patrick M. Madden (Acting).

Dated at Rockville, Maryland, this 31st day of July 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–19746 Filed 8–7–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles; Issue

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of issuance.

SUMMARY: The Nuclear Regulatory Commission (NRC) has issued Bulletin (BL) 2001–01 to all holders of operating licenses for pressurized-water nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel. BL 2001–01 addresses the recent discoveries of cracked and leaking reactor pressure vessel head (VHP) nozzles at several pressurized water reactors (PWRs) and the concerns raised about the structural integrity of VHP nozzles throughout the PWR industry. The purpose of the bulletin is to (1) request PWR licensees to provide information related to the structural integrity of the VHP nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements, and (2) require PWR licensees to provide to the NRC a written response in accordance with the provisions of 10 CFR 50.54(f). This information request is necessary to permit the assessment of plant-specific compliance with NRC regulations. The information will also be used by the NRC staff to determine the need for and to guide the development of additional regulatory actions to address cracking in VHP nozzles.

DATES: The bulletin was issued on August 3, 2001.

ADDRESSEES: Not applicable.

FOR FURTHER INFORMATION CONTACT: Allen L. Hiser, Jr., at 301–415–1034 or by e-mail to *alh@nrc.gov.*

SUPPLEMENTARY INFORMATION: Bulletin 2001–01 may be examined and/or copied for a fee at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and is accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* The ADAMS Accession No. for the bulletin is ML012080284.

If you do not have access to ADAMS or if there are problems in accessing documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 301–415–4737 or 1–800–397–4209, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 3rd day of August 2001.

For the Nuclear Regulatory Commission.

David B. Matthews,

Director, Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 01–19891 Filed 8–7–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[File No. 1–13862]

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration on the American Stock Exchange LLC; (Dia Met Minerals Ltd., Class A Subordinate Voting Shares, Without Par Value and Class B Multiple Voting Shares, Without Par Value)

August 1, 2001.

Dia Met Minerals Ltd., a British Columbia, Canada Corporation (''Issuer''), has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to Section 12(d) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Class A Subordinate Voting Shares, without par value and Class B Multiple Voting Shares (''Securities''), from listing and registration on the American Stock Exchange LLC (''Amex'').

The Issuer stated in its application that it has met the requirements of Amex Rule 18 by complying with all applicable laws in effect in the province of British Columbia, Canada, in which it is organized, and with the Amex's rules governing an issuer's voluntary withdrawal of a security from listing and registration.

In making the decision to withdraw the Securities from listing and registration on the Amex, the Issuer considered the cost associated with continued Amex listing and registration and decided that it is in the best interest of the shareholders to terminate its listing on the Amex. In addition, the Issuer represents that it has recently

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).