U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Private Fuel Storage (Independent Spent Fuel Storage Installation) Docket No. 72–22; Certified Review of LBP–01–03'' be held on February 14, and on less than one week's notice to the public.

By a vote of 5–0 on February 14, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Affirmation of Carolina Power & Light Company (Shearon Harris Nuclear Power Plant); Orange County's Petition for Review and Request for Immediate Suspension and Stay of the NRC Staff's 'No Significant Hazards Consideration' Determination and Issuance of License Amendment for Shearon Harris Spent Fuel Pool Expansion'' be held on February 14, and on less than one week's notice to the public.

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The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECY/smj/schedule.htm

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, D.C. 20555 (301–415–1969). In addition, distribution of this meeting notice over the Intenet system is available. if you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to dkw@nrc.gov.

Dated: February 15, 2001.

**Sandra M. Joosten,**

*Executive Assistant, Office of the Secretary.*

[FR Doc. 01–4369 Filed 2–16–01; 10:29 am]

**BILLING CODE 7590–01–M**

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## NUCLEAR REGULATORY COMMISSION

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

### I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 29, 2001, through February 9, 2001. The last biweekly notice was published on February 7, 2001.

### Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of 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, 11555 Rockville Pike, Room O–1F15, Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By March 23, 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, 11555 Rockville Pike, Room O–1F15, Rockville, Maryland, 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: Rules and Adjudications Branch, or may be delivered to the Commission's Public Document Room, 11555 Rockville Pike, Room O–1F15, Rockville, Maryland, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

*AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania*

*Date of amendment request:* January 15, 2001.

*Description of amendment request:* The proposed amendment revises the Technical Specification (TS) Design Features Section 5.4.2(f), ''Spent Fuel Storage,'' to remove the existing TS fuel assembly $U^{235}$ loading criterion for fuel assemblies stored in the spent fuel storage pool.

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

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change has no effect on the normal operating, design basis accident, or transient analyses applicable to the TMI [Three Mile Island] Unit 1 fuel storage requirements. Other existing TMI Unit 1 Technical Specification provisions ensure sub-criticality for normal and postulated accident conditions.

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

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Fuel assembly $U^{235}$ loading is not an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Discussion of fuel assembly $U^{235}$ loading in the TMI Unit 1 UFSAR [Updated Final Safety Analysis Report] ensures that changes to fuel designs that increase fuel reactivity relative to design assumptions for fuel storage are evaluated in accordance with the requirements of 10 CFR 50.59.

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

3. The proposed change does not involve a significant reduction in a margin of safety. The proposed change does not affect existing TMI Unit 1 Technical Specification requirements controlling maximum fuel enrichment, allowable enrichment vs. burnup, soluble boron requirements, storage rack spacing, allowable rack locations for fuel assembly storage or sub-criticality requirements for normal and accident conditions. These existing Technical Specification requirements ensure that the current margin of safety is not reduced. The fuel assembly $U^{235}$ loading criterion does not represent an input parameter or limiting design condition for any supporting design basis analyses applicable to the TMI Unit 1 spent fuel storage requirements.

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:* Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

*NRC Section Chief:* Marsha Gamberoni.

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

*Date of amendment request:* January 17, 2001.

*Description of amendment request:* The proposed amendments would relax Surveillance Requirement 3.6.1.3.7 by allowing a ''representative sample'' of excess flow check valves to be tested every 24 months, such that each excess flow check valve will be tested at least once every 10 years.

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

The current surveillance requirement frequency requires each reactor instrumentation line excess flow check valve to be tested every 24 months. The excess flow check valves at BSEP are designed to close automatically in the event of a line break downstream of the valve. The proposed change allows a reduction in the number of excess flow check valves to be tested every 24 months to approximately 20 percent of the valves each operating cycle. Industry operating experience demonstrates a high level of reliability for these excess flow check valves. A failure of an excess flow check valve to isolate cannot initiate previously evaluated accidents. Therefore, there is no increase in the probability of occurrence of an accident as a result of this proposed change. The postulated failure of an excess flow check valve to isolate is bounded by the limiting analysis in the Updated Final Safety Analysis Report (UFSAR). For a postulated break of an instrument line upstream of an excess flow check valve, leakage from the line rupture would be minimized by the line size or the flow-restricting orifice in the line. The rate and quantity of process fluid loss from an instrument line rupture is well within the capability of the reactor coolant make-up systems. The proposed change does not alter the design of the plants' instrument lines in any manner, and the integrity and functional performance of the secondary containment and Standby Gas Treatment system are not affected by this proposed change. The potential offsite radiological exposure associated with a postulated instrument line rupture upstream of an excess flow check valve is bounded by the main steam line break analysis and is substantially below the guidelines of 10 CFR 100. Therefore, the proposed license amendments do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change allows a reduced number of excess flow check valves to be tested each operating cycle. No other change in requirements are being proposed. Industry operating experience demonstrates the high reliability of the excess flow check valves. The potential failure of an excess flow check valve to isolate is bounded by the main steam line break analysis. The proposed license amendments do not physically alter the plants and will not alter the operation of the structures, systems, and components described in the UFSAR. Therefore, a new or different kind of accident will not be created.

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

Industry experience with excess flow check valves indicates that they have very low failure rates. The postulated failure of an excess flow check valve to isolate as a result of reduced testing is bounded by the limiting analysis in the UFSAR, which is the main steam line break analysis. For a postulated break of an instrument line upstream of an excess flow check valve, leakage from the line rupture would be minimized by the line size or the flow-restricting orifice in the line. The rate and quantity of process fluid loss from an instrument line rupture is well within the capability of the reactor coolant make-up systems. The proposed change does not alter the design of the plants' instrument line design in any manner, and the integrity and functional performance of the secondary containment and standby gas treatment system are not affected by this proposed change. Therefore, the proposed license amendments 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:* William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

*NRC Section Chief:* Richard P. Correia.

*Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois*

*Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois*

*Date of amendment request:* October 24, 2000.

*Description of amendment request:* The proposed amendment would revise the technical specifications to change the Westinghouse references for Best Estimate Large Break Loss of Coolant Accident (LOCA) analysis methodology. Reanalysis of large break LOCA transients, utilizing the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC, was performed to demonstrate that 10 CFR 50.46 acceptance criteria are satisfied at uprated power conditions.

*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?

No physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. The proposed TS changes simply involve updating the references in TS 5.6.5.b, ''Core Operating Limits Report (COLR),'' to reference the Westinghouse Best Estimate Large Break LOCA analysis methodology (i.e., Westinghouse topical report, WCAP–12945–P–A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, ''Code Qualification Document for Best Estimate LOCA Analysis,'' March 1998). The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant; therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, ''Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.'' Furthermore, the re-performance of the Large Break LOCA analysis has no effect on the performance of the ECCS equipment. No other accident consequence is potentially affected by this change.

All systems will continue to be operated in accordance with current design requirements under the new analysis, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Protection System (RPS) or Engineered Safety Features (ESF) setpoints because of the new analysis methodology.

Based on the analysis, it is concluded that the proposed TS 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 accident from any accident previously evaluated?

There are no physical changes being made to the plant as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. No new modes of plant operation are being introduced. The configuration, operation and accident response of the Byron Station and the Braidwood Station systems, structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the Westinghouse Best Estimate Large Break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any RPS or ESF actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

It has been shown that the analytic technique used in the Westinghouse Best Estimate Large Break LOCA analysis methodology realistically describes the expected behavior of the Byron Station and Braidwood Station reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs have been evaluated. The analysis has demonstrated that there is a high probability that all acceptance criteria contained in 10 CFR 50.46, paragraph b, continues to be satisfied. Based on this review, the proposed TS 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 requested amendments involve no significant hazards consideration.

*Attorney for licensee:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767

*NRC Section Chief:* Anthony J. Mendiola.

*Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois*

*Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois*

*Date of amendment request:* November 7, 2000.

*Description of amendment request:* The proposed amendment would revise the technical specifications to extend the TS Surveillance Test Interval (STI) from a 92-day STI to an 18-month STI, for the Solid State Protection System (SSPS) slave relay types that meet the acceptance criteria for the reliability assessments performed in accordance with the methodology described in the NRC approved Westinghouse Electric Corporation Topical Reports.

*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 changes are consistent with the NRC approved Westinghouse Electric Corporation Topical Reports, WCAP–13877, ''Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays,'' WCAP–13878, ''Reliability Assessment of Potter & Brumfield MDR Series Relays,'' and WCAP–13900, ''Extension of Slave Relay Surveillance Testing Intervals,'' that analyze extending the Solid State Protection System (SSPS) slave relay surveillance test interval (STI) for the Westinghouse Type AR slave relays and for the Potter & Brumfield MDR Series slave relays. The reliability assessment of the slave relays was comprised of a failure modes and effects analysis (FMEA) and an aging assessment of the slave relays. WCAP–13877 and WCAP–13878 verified that the Westinghouse Type AR and the Potter & Brumfield MDR Series slave relays are highly reliable and that degradation of the slave relays is sufficiently slow (i.e., the time to failure due to degradation is sufficiently long) that an 18-month STI will adequately identify slave relay failures. A 92-day STI is no more likely to detect significant changes in the SSPS slave relays than an 18-month STI. The results demonstrate that extending the SSPS slave relay STI from 92 days to 18 months does not adversely affect the reliability of the SSPS slave relays utilized in Engineered Safety Features Actuation System (ESFAS) functions.

The high reliability of these slave relays precludes the need for more frequent periodic surveillance testing to verify operability.

As stated in WCAP–13877 and WCAP–13878, the overly conservative 92-day STI can be extended to an 18-month STI without impact or consequence to slave relay reliability. In addition, the proposed changes will not adversely affect the ability of the SSPS to perform its safety function. The same ESFAS instrumentation is being used and the ESFAS reliability is being maintained with the proposed changes. Because the reliability of the slave relays used in the ESFAS applications is so high, elimination of the routine surveillance testing of the slave relays when the reactor is at power will have a positive impact on ESFAS availability and plant safety. The proposed changes will not modify any system interface and will not increase the likelihood of any accident initiator because such events are independent of the proposed changes. Therefore, the probability of an accident previously evaluated is not increased.

The proposed changes will not modify, degrade, or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of any accident described in the Updated Final Safety Analysis Report (UFSAR). The ESFAS instrumentation remains capable of performing its intended safety function of mitigation of consequences of accidents or transients. Therefore, the consequences of an accident previously evaluated are not 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 accident from any accident previously evaluated?

The proposed changes do not alter the performance of the ESFAS. The proposed changes only extend the STI, and no changes to the testing methodology or the way in which the slave relays are tested are being proposed. No new equipment is being installed, and no installed equipment is being operated in a new or different manner with the proposed changes. Extending the STI will maintain the reliability of the slave relays as demonstrated by the NRC approved FMEA and aging assessment, and may improve the reliability of the system by reducing potential test-induced degradation. As documented in WCAP–13877 and WCAP–13878, an STI of 92 days is no more likely to detect significant changes in the SSPS Type AR and MDR Series slave relays than a STI of 18 months.

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

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

The proposed changes do not affect the total ESFAS response assumed in the safety analysis. The periodic slave relay functional verification is relaxed because of the demonstrated high reliability of the slave relays and their insensitivity to any short-term wear or aging effects. The Westinghouse Owners Group (WOG) program to extend the STI for the slave relays, as documented in the NRC approved WCAP–13877 and WCAP–13878, has concluded that the slave relays used in the SSPS are highly reliable and that the surveillance testing at a frequency of 18 months, instead of the 92-day STI currently required, does not significantly decrease any margin of safety assumed in the safety analysis. Plant safety will be improved by limiting the amount of on-line testing that will be performed, because on-line testing of the slave relays results in the removal of a train of equipment from service or manipulation of specific safety-related equipment which is then no longer able to perform its safety function if called upon until the surveillance test is completed. The proposed changes also act to improve plant safety by reducing equipment degradation and reducing unnecessary burden on the operating personnel. There are no changes in testing methodology or performance criteria.

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 50.929c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

*Attorney for licensee:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60609–0767.

*NRC Section Chief:* Anthony J. Mendiola.

*Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois*

*Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois*

*Date of amendment request:* November 13, 2000.

*Description of amendment request:* The proposed amendment would revise the technical specifications to delete the ''Power Range Neutron Flux High Negative Rate,'' Trip Function from Reactor Trip System Instrumentation. The proposed change allows elimination of this unnecessary function and thereby reduces the potential for a transient. The proposed changes are consistent with the Westinghouse Topical report previously accepted by the NRC.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of he 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 removal of the Power Range Neutron Flux High Negative Rate Trip (i.e., Negative Flux Rate Trip (NFRT)) Function does not increase the probability or consequences of reactor core damage accidents resulting from dropper Rod Cluster Control Assembly (RCCA) events previously analyzed. The safety functions of other safety related systems and components, which are related to mitigation of these events, have not been altered. All other primary Reactor Trip System (RTS) and Engineered Safety Features Actuation Systems (ESFAS) protection functions are not impacted by the elimination of the NFRT Function. The NFRT circuitry detects and responds to negative reactivity insertion due to RCCA misoperation events should they occur. Therefore, the NFRT Function is not assumed in the initiation of such events. Because the NFRT Function is being eliminated from the plant, it can no longer actuate and cause a transient. The consequences of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) are unaffected by the proposed changes because no change to any equipment response or accident mitigation scenario has resulted.

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 accident from any accident previously evaluated?

The deletion of the NFRT Function does not create the possibility of a new or different kind of accident than any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of any safety related systems. It has been demonstrated that the NFRT Function can be eliminated by the NRC approved methodology described in Westinghouse Topical Report WCAP–11394–P–A, ''Methodology for the Analysis of the Dropped Rod Event,'' dated January 1990. The Braidwood Station and the Byron Station cycle-specific analyses have confirmed that for a dropped RCCA(s) event, no direct reactor trip or automatic power reduction is required to meet the Departure From Nucleate Boiling (DNB) limits for this Condition II, ''Faults of Moderate Frequency,'' event. The NFRT Function is not credited either as a primary or backup mitigation feature for any other UFSAR event. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety associated with the licensing basis acceptance criteria for any postulated accident is unchanged. It has been demonstrated that the NFRT Function can be eliminated by the NRC approved methodology described in WCAP 11394–P–A. The Braidwood Station and the Byron Station cycle-specific analyses have confirmed that for a dropped RCCA(s) event, DNB limits are not exceeded with the proposed changes. Conformance to our licensing basis acceptance criteria for Design Basis Accidents (DBAs) and transients with the deletion of the NFRT Function is demonstrated, and DNB limits are not exceeded. The proposed changes will have no adverse effect on the availability, operability, or performance of the safety related systems and components assumed to actuate in the event of a DBA or transient. 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:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendment request:* February 29, 2000.

*Description of amendment request:* The proposed amendment would reduce the number of safety valves required for overpressure protection at Dresden, Unit 2, by removing from Technical Specifications (TS) Section 3.6.E, the safety valve function of the Target Rock safety/relief valve (SRV). The proposed amendment would move the safety valve lift pressure setpoints from TS Section 3.6.E to TS Section 4.6.E, remove the Target Rock SRV setpoints from TS, and change the number of safety valves from nine to eight. The proposed amendment would also remove footnote ''c'' of Unit 3, TS Section 4.6.E, since this footnote was only applicable to Unit 3, Cycle 15 which has been completed.

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established, consistent with Nuclear Regulatory Commission (NRC) approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change to reduce the number of required safety valves from nine (9) to eight (8) does not affect the ability of plant systems to adequately mitigate the consequences of an accident previously evaluated.

This conclusion was derived by evaluating all applicable analyses including thermal limit, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) pressurization events, margin to unpiped safety valve, anticipated transient analysis without scram events, Loss Of Coolant Accident (LOCA), station blackout, and 10 CFR 50, Appendix R analyses. Therefore, there is no increase in the probability or consequences of an accident previously evaluated because the analyses supports operation without crediting the Target Rock Safety Relief Valve safety mode function.

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

The requested change has been previously evaluated by evaluating all applicable analyses including thermal limit, ASME B&PV pressurization events, margin to unpiped safety valve, anticipated transient analysis without scram events, station blackout, LOCA, and 10 CFR 50, Appendix R analyses. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the analyses support operation without crediting the Target Rock safety relief valve safety function. No new failure modes will be introduced upon implementation of the proposed changes, therefore, the possibility of a new and different accident has not been created.

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

Changing the required number of safety valves from nine (9) to eight (8) will not involve any reduction in margin of safety. This conclusion was derived by evaluating all existing analyses including thermal limit, ASME B&PV pressurization events, margin to unpiped safety valve, anticipated transient analysis without scram events, station blackout, LOCA, and 10 CFR 50, Appendix R analyses. The analyses previously evaluated remain valid, therefore, a significant reduction in the margin of safety does not exist.

Therefore, based upon the above evaluation, ComEd has concluded that these changes do not constitute a significant hazards consideration.

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

*Attorney for licensee:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

*NRC Section Chief:* Anthony J. Mendiola.

*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:* February 14, 2000.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TSs) to correct various editorial errors and make other administrative changes. Specifically, the proposed amendment would make administrative changes that revise: (a) Tables 3.6–1 and 4.4–1 to correct listing and editorial errors, (b) TS 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv), (c) Figures 3.10–2 through 3.10–6 to remove these figures, (d) Table 4.1–1 to reflect change in level indication components, (e) TS 4.19.B and 6.14.1.1 to correct editorial errors, (f) TS 6.12.1 to reference the current sections of 10 CFR Part 20, (g) TS 6.12.1 to reflect an organizational title change, and (h) TS 6.13.2 to correct a typographical error.

*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) Changes To Tables 3.6–1 And 4.4–1 To Correct Listing And Editorial Errors

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed changes are administrative in nature. The changes involve correcting errors in Table 3.6–1 and additions to Tables 3.6–1 and 4.4–1 to reflect UFSAR [Updated Final Safety Analysis Report] Table 5.2–1 and the IST [inservice testing] Program. These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed changes. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

(b) Change To Section 3.8.B.10 To Reflect The Wording In 10 CFR 50.54(m)(2)(iv)

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves updating Section 3.8.B.10 to reflect 10 CFR 50.54(m)(2)(iv). This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(c) Deletion Of Figures 3.10–2 Through 3.10–6

(1) Does the proposed license amendment involve a significant increase in the

probability or in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves the deletion of Figures 3.10–2, 3.10–3, 3.10–4, 3.10–5 and 3.10–6. This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(d) Change To Table 4.1–1 To Reflect Change In Level Indication Components

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. This change does not affect possible initiating events for accidents previously evaluated or operation of the facility. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR [control room]). The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The safety analysis of the facility remains complete and accurate. The plant conditions for which the design basis accidents have been evaluated are still valid. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses [sic] sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR). Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR). Also, there are no changes to the operation of the facility. Thus the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(e) Change To Sections 4.19.B And 6.14.1.1 To Correct Editorial Errors

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed changes are administrative in nature. The change in Sections 4.19.B and 6.14.1.1 involve amending ‘‘the Semiannual Radioactive Effluent Release Report’’ to ‘‘the Annual Radioactive Effluent Release Report.’’ These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

(f) Change To Section 6.12.1 To Reference The Current Sections Of 10 CFR [Part] 20

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves updating Section 6.12.1 to reference 10 CFR 20.1601(a) and 10 CFR 20.1601(b). This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(g) Change To Section 6.12.1 To Reflect An Organizational Title Change

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves updating Section 6.12.1 to use the title ‘‘Shift Manager’’ instead of ‘‘Senior Watch Supervisor.’’ This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of

accident from any accident previously evaluated?

No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(h) Change To Section 6.13.2 To Correct A Typographical Error

(1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. The proposed change is administrative in nature. The change involves updating Section 6.13.2 from ''DOR [Division of Operating Reactors] Guidelines of NUREG–0588'' to ''DOR Guidelines or NUREG–0588.'' This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. 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:* Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

*NRC Section Chief:* Marsha Gamberoni.

*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:* December 11, 2000.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TSs) to provide editorial revisions, clarifications, and corrections. Specifically, the proposed amendment would: (1) Provide updated information and corrections to the TS cover page, table of contents, and list of figures, (2) revise TS 4.5.E, ''Control Room Air Filtration System,'' to remove an incorrect system test description and provide consistent test values for system flow rate and filter efficiency, (3) revise TS 6.2.1.a, ''Facility Management and Technical Support,'' to reference the Quality Assurance Program Description as the location of the documentation rather than the Updated Final Safety Analysis Report, (4) revise TS 6.9.1.7, ''Monthly Operating Report,'' to change the recipient of the Monthly Operating Report, and (5) correct the periodicity of the Radioactive Effluent Release Report from annual to semiannual in TS 6.15, ''Offsite Dose Calculation Manual'' and TS 6.16, ''Major Changes to Radioactive Liquid, Gaseous and Solid Waste Systems.'' In addition, the proposed change revises TS Figure 5.1–1B concerning the indicated vent location associated with Indian Point Unit 3 (IP3). The labels for the plant vent and the machine shop are reversed.

*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 license amendment involve a significant increase in the probability or [ * * * ] consequences of an accident previously evaluated?

The proposed changes consist of editorial changes, administrative changes, clarifications, and corrections to existing TSs. These changes do not involve a change to the design or operation of any plant system nor are any of the safety analyses affected as a result of these changes. Accordingly, the initiators of any accident as well as any system relied upon for the mitigation of an accident are not affected by the proposed changes. Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed changes do not involve a change to the design or operation of any plant system. These changes include editorial changes, administrative changes, clarifications, and corrections of existing TSs and, therefore, do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes consist of editorial changes, administrative changes, and clarifications to existing TSs and do not involve changes to any margin of safety.

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

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

*NRC Section Chief:* Marsha Gamberoni.

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

*Date of amendment request:* January 8, 2001.

*Description of amendment request:* The proposed change revises the lower limit of the allowable containment internal pressure in Technical Specification (TS) 3.6.1.4, ''Containment Systems—Internal Pressure,'' from 14.375 pounds per square inch, absolute (psia) to 14.275 psia.

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

Response: The proposed change revises the lower limit of the allowable containment internal pressure in TS 3.6.1.4 from 14.375 to 14.275 psia. This change will allow additional operating margin for the containment atmosphere purge (CAP) system during conditions of low atmospheric pressure. The containment minimum pressure parameter is not an accident initiator and does not affect the probability of any initiating event scenario. Although the TSs will allow a lower initial containment internal pressure, the current analyses for the associated design events are not affected since the lower pressure has already been conservatively included. The proposed change in initial containment internal pressure is bounded in the current design. Therefore, this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. Will the 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?

Response: The proposed change affects the TS allowed lower limit on containment internal pressure and consequently the atmospheric range in which the CAP system can be operated. The change in the lower limit on containment internal pressure is encompassed by current design analyses and does not result in a change of analyzed conditions or analyzed operating ranges.

Based on the proposed TS change, CAP system operation will be allowed at a lower atmospheric pressure. This change does not change the function of the system or its method of operation. Although the initial atmospheric pressure at which the CAP system can be initiated is being lowered, this is within the current design of the CAP system and does not change the differential pressures at which it will be operated.

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

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

Response: The proposed change makes use of the initial containment pressure assumption values used in the current analyses to provide additional operating margin for the CAP system. The margin of safety that was inherent in the results of these safety analyses has been preserved. The associated analyses ensure the negative pressure differential associated with an inadvertent actuation of the containment spray system is acceptable, and ensure that the emergency core cooling system can satisfy its design safety function under worst case conditions. The calculated maximum differential pressure is 0.49 psid [pounds per square inch differential] which is within the design limit of 0.65 psid. The peak clad temperature for the worst case large break loss of coolant accident is 2177°F which is within the acceptance criteria given in 10CFR50.46. Since the proposed change does not affect the initial containment pressure utilized in these analyses, the results of the analyses are unchanged. Therefore, there is no affect on any margin of safety associated with this parameter.

Based on the above No Significant Hazards Consideration Determination, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92; (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC final environmental statement.

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

*Date of amendment request:* September 1, 2000.

*Description of amendment request:* The proposed amendments would revise the technical specifications to add a new requirement for the Main Steam Line Radiation Monitor mechanical vacuum pump trip function.

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

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

The addition of the MSLRM [Main Steam Line Radiation Monitor] automatic trip signal to the MVP [mechanical vacuum pump] has no adverse effect on safety. The addition of Surveillance Requirements (SRs) and the Limiting Condition for Operation (LCO) to our TS enhances current safety features of the plant by establishing controls for a required, and currently functional, safety feature. The automatic trip function of the MVP does not serve as an initiator for any accidents evaluated in Chapter 15, ''Accident and Transient Analysis,'' of the Updated Final Safety Analysis Report. Therefore, this change will not result in an increase of either the probability or consequences of an accident.

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

These proposed changes involve the addition of the MVP trip input from the Main Steam Line Tunnel High Radiation signal. The addition of this function does not represent a change in operating parameters or equipment configuration for DNPS [Dresden Nuclear Power Station], Units 2 and 3. Operation of DNPS, Units 2 and 3, under the proposed changes does not create the possibility of a new or different type of accident previously evaluated.

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

These proposed changes create a TS LCO and identify SRs for the MVP trip input from the MSLRM signal. Operation under the proposed change will not change any plant operation parameters, nor any protective system setpoints. The calculations of off site dose demonstrate that with the MVP trip instrumentation operating properly, the doses that result from a CRDA [control rod drop accident] with the MVP operating are well within 10 CFR Part 100, ''Reactor Site Criteria,'' limits. [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:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

*NRC Section Chief:* Anthony J. Mendiola.

*Exelon Generating Company, LLC (Exelon), Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania*

*Date of amendment request:* November 20, 2000.

*Description of amendment request:* PECO Energy Company (PECO) proposed changes to the Technical Specifications (TSs) that would revise the heatup, cooldown, and inservice test Pressure-Temperature (P–T) limitations (TS Figure 3.4.6.1–1) of the Limerick Generating Station (LGS), Unit 2, Reactor Pressure Vessel (RPV) to a maximum of 32 Effective Full Power Years (EFPY). In addition, the licensee proposed text changes to both Limiting Condition for Operation 3.4.6.1 and Surveillance Requirement 4.4.6.1.1 to delete the reference to the A' curve on TS Figure 3.4.6.1–1 since this curve will not be included in the proposed Figure 3.4.6.1–1. The licensee also proposed adding an intermediate hydrotest curve (Curve $A_{22}$) to TS Figure 3.4.6.1–1, which is valid to 22 EFPY. By letter dated January 30, 2001, Exelon stated that it has assumed responsibility, as of the date of the transfer, for the active items on the Limerick Units 1 and 2 dockets previously submitted by PECO, including the subject amendment request.

Moreover, Exelon is revising its TS Bases Section B 3/4.4.6 to update several RPV material chemistry parameters. The licensee identified the need for these revisions during a Certified Material Test Report data search performed by General Electric Company during development of the proposed P–T curves.

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

1. 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 P–T curves. The proposed changes do not modify the reactor coolant pressure boundary, i.e., there are no changes in operating pressure, materials or seismic loading. The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. The proposed P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, in conjunction with ASME Code Case N–640. The proposed P–T curves were established in compliance with the methodology used to calculate the predicted irradiation effects on vessel beltline materials. Usage of these procedures provides compliance with the intent of 10 CFR 50, Appendix G, and provides margins of safety that ensure that failure of the reactor vessel will not occur. The proposed P–T curves prohibit operational conditions in which brittle fracture of reactor vessel materials is possible. Consequently, the primary coolant pressure boundary integrity will be maintained. Therefore, the proposed changes do not involve a significant increase in the probability or 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.

The proposed changes to the P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, and ASME B&PV Code, Section XI, Appendix G, in conjunction with ASME Code Case N–640. Compliance with the proposed P–T curves will ensure that conditions in which brittle fracture of primary coolant pressure boundary materials are possible will be avoided. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any new failure mode from previously evaluated accidents. Further, the proposed changes to the P–T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate nor 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. The proposed TS changes do not involve a significant reduction in a margin of safety.

The proposed changes reflect an update of the P–T curves to extend the reactor pressure vessel operating limit to 32 Effective Full Power Years (EFPY). The revised curves are based on the latest ASME guidance. The revised P–T limits, which provide more operational flexibility than the current limits, were established in accordance with current regulations and the latest ASME Code information. No plant safety limits, set points, or design parameters are adversely affected by the proposed TS changes. These proposed changes maintain the relative margin of safety commensurate with that which existed at the time that the ASME B&PV Code, Section XI, Appendix G, was approved in 1974.

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

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

*Attorney for licensee:* J.W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Exelon Generating Company, 2301 Market Street, Philadelphia, PA 19101

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* January 18, 2001.

*Description of amendment request:* Exelon requested a Technical Specification (TS) change which will revise Surveillance Requirement (SR) 4.9.2.d.1 to clarify that ''shorting links'' do not need to be removed, if adequate shutdown margin has been demonstrated, when moving a control rod during Operational Condition 5.

*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.

This TS Change Request revises SR 4.9.2.d.1 to clarify that ''shorting links'' do not need to be removed if adequate shutdown margin has been demonstrated when a control rod is withdrawn during Operational Condition 5. This revision ensures that the words and intent of the SR 4.9.2.d.1 match the words and intent of Limiting Condition for Operation (LCO) 3.9.2.d, and will improve the readability of the SR for plant operators. This change to SR 4.9.2.d.1 will clarify that ''shorting links'' can remain installed if adequate shutdown margin has been demonstrated any time a control rod is withdrawn in Operational Condition 5. This revision does not impact any accident or transient events. There are no new initiators created by this revision. Additionally, this revision will not impact any existing analyses or requirements contained in the Updated Final Safety Analysis Report. No changes in the operation of the plant during power operation or refueling will occur as a result of this revision. Therefore, the proposed TS revision does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed TS revision will not impact any physical changes to plant structures, systems, or components. The design, function, and reliability of the Reactor Protection System will not be impacted by this revision. This revision does not adversely impact any equipment which is required for the prevention or mitigation of accidents or transients. This revision ensures that the words and intent of the SR 4.9.2.d.1 match the words and intent of LCO 3.9.2.d, and will improve the readability of the SR for plant operators. 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 TS change does not involve a significant reduction in a margin of safety.

This proposed revision to SR 4.9.2.d.1 does not affect any safety limits or analytical limits. There are also no changes to accident or transient core thermal hydraulic conditions, minimum combustible concentration limits, or fuel or reactor coolant boundary design limits, as a result of this proposed change. This revision ensures that the words and intent of the SR 4.9.2.d.1 match the words and intent of LCO 3.9.2.d. Therefore, the 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:* J. W. Durham, Sr., Esquire, Sr. V.P. and General

Counsel, Exelon Generating Company, 2301 Market Street, Philadelphia, PA 19101.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* January 18, 2001.

*Description of amendment request:* Exelon requested a Technical Specification (TS) change which will revise the Units 1 and 2 TS Table 1.2, ''Operational Conditions,'' to allow placing the reactor mode switch to the REFUEL position during Operational Conditions 3 and 4 to accommodate post maintenance and surveillance testing on control rod drives.

*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 proposed revision allows a single control rod to be withdrawn under control of the reactor mode switch REFUEL position and the one-rod-out interlock in Operational Conditions 3 and 4. This change does not affect any existing accident initiators. There is no change to the coupling integrity of the control rod during this accident. Although this change would allow an increase in the frequency of single control rod withdrawals in Operational Conditions 3 and 4, the probability of the previously analyzed accidents is not affected.

The onsite and offsite radiological consequences of previously analyzed accidents in Operational Conditions 3 and 4 are not affected by this proposed change. This change does not affect any existing accident mitigators. The shutdown margin combined with the refueling interlocks prevent a rod withdrawal error while in refueling thereby preventing inadvertent criticality. There is no impact on the ability of the Reactor Protection System (RPS) circuitry to mitigate a Control Rod Drop Accident as described in the Safety Analysis Report, nor is there an increase in the number of fuel failures from this accident. As a result, the probability and consequences of previously analyzed accidents are not significantly increased.

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

There are no new accident initiators created by the proposed revision to Table 1.2. Single control rods can be withdrawn in Operational Conditions 3 and 4 under the existing Technical Specifications to permit control rod recoupling. The proposed revision would expand this provision to other control rod maintenance and testing activities performed in Operational Conditions 3 and 4. The withdrawal of individual control rods in Operational Conditions 3 and 4 is a mode of operation permitted under limited circumstances by the existing TSs. The additional control rod maintenance and testing activities which could be performed in Operational Conditions 3 and 4, are already permitted by the existing TSs in Operational Conditions 1, 2, 4, and 5.

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

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

The TSs currently permit single control rod withdrawal for the purpose of control rod recoupling when in Operational Conditions 3 or 4 if the one-rod-out interlock is Operable. This change allows additional activities for which a single control rod may be withdrawn when in Operational Conditions 3 or 4, with the same restriction that the one-rod-out interlock be Operable.

The operability requirements for the one-rod-out interlock and the shutdown margin requirements of TS 3.1.1 ensure the reactor will be maintained subcritical during single control rod withdrawals. Therefore, this change will 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:* J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Exelon Generation Company, 2301 Market Street, Philadelphia, PA 19101.

*NRC Section Chief:* James W. Clifford.

*Exelon Generating Company, LLC (Exelon), Docket No. 50–353, Limerick Generating Station, Unit 2, Montgomery County, Pennsylvania.*

*Date of amendment request:* February 1, 2001.

*Description of amendment request:* Exelon proposed changes that would revise Technical Specification (TS) 2.1 to incorporate revised Safety Limit Minimum Critical Power Ratios due to the cycle-specific analysis performed by Global Nuclear Fuel for Limerick Generating Station, Unit 2, Cycle 7, which will include the use of the GE–14 fuel product line.

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

The derivation of the cycle specific Safety Limit Minimum Critical Power Ratios (SLMCPRs) for incorporation into the Technical Specifications (TS), and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Amendment 25 was approved by the NRC [Nuclear Regulatory Commission] in a March 11, 1999 safety evaluation report.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling. The GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of this change. Therefore, the 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 SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLMCPRs are calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Additionally, the GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000, which provides the fuel licensing acceptance criteria. The SLMCPR is not an accident initiator, and its revision will 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 a margin of safety.

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLMCPRs, which includes the use of GE–14 fuel. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change will not involve a significant reduction in the margin of safety previously approved by the NRC.

Based on the staff's review of the licensee's evaluation, 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:* J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Exelon Generating Company, 2301 Market Street, Philadelphia, PA 19101.

*NRC Section Chief:* James W. Clifford.

*Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin*

*Date of amendment request:* January 13, 2000.

*Description of amendment request:* The proposed amendment would change the Kewaunee Nuclear Power Plant Technical Specification 3.6, ''Containment.'' The proposed amendment would add limiting condition for operation and allowed outage times for containment penetrations and associated isolation devices to provide clear guidance. Also, the proposed amendment would provide additional information, clarification, and uniformity to the basis of the associated technical specification.

*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.

This Technical Specification [TS] change provides definition for the [Allowable Outage Time] AOT for a containment isolation valve and containment air lock. The original design and design basis of the plant is still maintained and the probability and consequences of previously evaluated accidents is unchanged. In our current Technical Specifications the allowed outage time for a safeguards 480-volt bus is 24 hours. The basis for this outage time states:

''The intent of this TS is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safety features equipment following an accident.''

With one 480-volt safeguards bus out of service an associated motor operated containment isolation valve is also out of service. Since the 24-hour AOT is part of Kewaunee's original design basis, allowing the containment isolation valves to be out of service for 24 hours does not increase the probability or consequences of an accident previously evaluated.

A risk assessment of the probability of a -loss-of-coolant-accident with a train of containment isolation failing during a 4-hour verse a 24-hour time span was conducted. The probability of [loss-of-coolant accident] LOCA coincident with the failure of containment isolation occurring during a 4-hour period versus a 24-hour period is shown on Figure 1[ in licensee's submittal]. This change in probability is considered insignificant.

The proposed TS changes do not involve any physical or operational changes to structures, systems or components. The current safety analysis and design basis for the accident mitigation functions of the containment, the airlocks, and the containment isolation valves are maintained. On-site and off-site dose consequences remain unaffected.

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

The function of the containment vessel is to contain the radiologically hazardous material following a LOCA. By maintaining at least one containment isolation barrier intact the vessel can perform its function. This amendment still ensures that at least one barrier is intact or the leakage is evaluated not to exceed that which is already evaluated and allowed by current technical specification.

The accidents considered are found in the Safety Analysis, Section 14 of the [Updated Safety Analysis Report] USAR. The proposed change does not involve a change to the plant design (structures, systems or components) or operation. No new failure mechanisms beyond those already considered in the current plant Safety Analysis are introduced. No new accident is introduced and no safety-related equipment or safety functions are altered. The proposed change does not affect any of the parameters or conditions that contribute to initiation of any accidents.

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

With one containment barrier intact during plant operation the isolation of containment is still ensured. The plant's original design basis addressed the inability of one of the two containment isolation valves to operate for a 24-hour period. As this AOT has been previously considered, there therefore is no reduction in a margin of safety.

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

*Attorney for licensee:* Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

*NRC Section Chief:* Claudia M. Craig.

*Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin*

*Date of amendment request:* January 18, 2001.

*Description of amendment request:* The proposed amendment would change the Kewaunee Nuclear Power Plant Technical Specification 3.10.m for reactor coolant minimum flow from the current value of 85,500 gallons per minute (gpm) to 93,000 gpm due to the replacement of steam generators scheduled for the fall 2001.

*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 change in Reactor Coolant Minimum Flow value for TS 3.10.m proposed in this amendment request is needed to reflect operating characteristics of the new [Replacement Steam Generators] RSGs. Accident analyses affected by the RSGs have each been evaluated to establish that there is no significant change in the documented results (Attachment 3). These evaluations have shown that the proposed value for Reactor Coolant Minimum Flow is bounded by the Thermal Design Flow value used in the analyses and provides greater margin to safety analysis acceptance criteria (e.g., [Departure from Nucleate Boiling] DNB). All safety analysis acceptance criteria are satisfied. Since Reactor Coolant flow values for the RSG conform to the design bases and are bounded by the existing safety analyses, changing the technical specification within limits of the bounding accident analyses will not cause an 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 proposed change is fully consistent with current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Thus, it does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the manner in which Safety Limits, Limiting

Safety System Setpoints, or Limiting Conditions for Operation are determined. It returns TS 3.10.m for Reactor Coolant Minimum Flow to a value slightly higher, thus more conservative, than the value specified for the [Original Steam Generators] OSG when new. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation. Analysis of the effect of the proposed Reactor Coolant Minimum Flow limitation on [Loss-of-Coolant-Accident] LOCA and non-LOCA transients determined that all safety analysis acceptance criteria are satisfied at a [Thermal Design Flow] TDF that bounds the revised Reactor Coolant Minimum Flow and all [Kewaunee Nuclear Power Plant] KNPP safety requirements continue to be met. 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:* Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

*NRC Section Chief:* Claudia M. Craig.

*Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin*

*Date of amendment request:* February 5, 2001.

*Description of amendment request:* The proposed amendment would change the Kewaunee Nuclear Power Plant Technical Specification 3.1.d.2 to reduce the maximum allowable leakage of primary system reactor coolant to the secondary system from 500 gallons per day (gpd) through any one steam generator to 150 gpd through any one steam generator. In addition, the proposed amendment would remove reference to voltage based repair criteria.

*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 change in Leakage of Reactor Coolant value proposed by this request for [Technical Specification] TS 3.1.d.2 complies with [Nuclear Energy Institute] NEI 97–06, ''Steam Generator Program Guidelines.'' [Nuclear Management Company, LLC] NMC evaluated accident analyses affected by [steam generator] SG tube leakage and determined that this change continues to be bounded by the existing licensing and design basis. Design basis accidents and transients, including steam generator tube rupture (SGTR), were analyzed using Westinghouse Model 54F steam generator assumptions as part of steam generator replacement. These evaluations show that the proposed 150 gpd [gallons per day] value for Leakage of Reactor Coolant is bounded by the larger value used in applicable existing design basis accident and transient analyses. The 150 gpd leak rate provides increased margin to acceptance criteria found in these analyses. All acceptance criteria are satisfied and SG primary to secondary leakage values for the [replacement steam generator] RSG conform to the existing design bases and are bounded by the existing safety analyses. Changing the technical specification within limits of the bounding accident analyses cannot change the probability or consequence of an accident previously evaluated. Removal of an allowance for voltage-based alternate repair criteria defaults to a more conservative repair criteria. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

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

The 150 gpd value proposed for maximum allowable Leakage of Reactor Coolant is consistent with current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. The proposed value for maximum allowable leakage through any one steam generator is bounded by currently licensed design basis accident and transient analyses of record. Removal of a reference in the TS to voltage-based repair criteria leaves in its place a more conservative, more restrictive criteria for repair or plugging of steam generator tubes. Thus, this proposal does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. It sets TS 3.1.d.2 for Leakage of Reactor Coolant to a lower, thus more conservative, value than that previously specified and approved for use by the NRC [Nuclear Regulatory Commission]. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Analyses of applicable transients were performed using a primary to secondary leakage rate greater than the rate proposed by this request. All safety analysis acceptance criteria are satisfied at this value and all [Kewaunee Nuclear Power Plant] KNPP safety requirements continue to be met. The 150 gpd leak rate proposed by this amendment request is bounded by these analyses. Removal of reference to use of voltage-based repair criteria from TS 3.1.d.2 and its basis leaves an existing and more conservative repair criteria in place. Thus, changes proposed by this request 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:* Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

*NRC Section Chief:* Claudia M. Craig.

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

*Date of amendment requests:* January 11, 2001.

*Description of amendment requests:* The proposed amendment deletes requirements from the Technical Specifications (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 technical specifications (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 January 11, 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:* Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

*NRC Section Chief:* Stephen Dembek.

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

*Date of application for amendment:* January 18, 2001.

*Brief description of amendment request:* The amendment request identifies an unreviewed safety question related to the planned replacement of the engineered safety features (ESF) transformers with new transformers having active automatic load tap changers (LTCs). Markups to the Final Safety Analysis Report (FSAR) were included in the application.

*Basis for proposed no significant hazards 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?

Based on the review of the modification details there is an insignificant increase in the probability of a malfunction of equipment important to safety, however there is no increase in the probability of an accident previously evaluated. The modification has no effect on the radiological consequences of accidents previously evaluated. Installation of the LTCs does not impact accident initiators though a failure mode has been identified that can increase the probability of malfunction, a risk study shows this risk is insignificant.

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

The overall effect of the malfunction of the LTC controllers would lead to a loss of the associated ESF bus which is not a new failure mode that can lead to a new or different kind of accident than previously evaluated. The LTC failure effects are limited to the associated ESF train, therefore this type of failure meets the definition of a single failure as defined in 10 CFR 50 Appendix A for operation under normal (Non T/S [technical specification] action) conditions. Additionally, during the 10 CFR 50.59 evaluation for the LTCs criteria (a)(2)(ii) with respect to accidents of a different type was not met.

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

The installation of the replacement transformers with load tap changers will help assure the required minimum NB bus voltage established by Reference 7.10 [design calculations] under a wider variation of grid voltage.

Current Technical Specification Bases for the offsite power distribution system are covered in sections B3.8.1–AC Sources-Operating, B3.8.9–Distribution Systems-Operating, B3.8.2–AC Sources-Shutdown, and B3.8.10–Distribution Systems-Shutdown. These bases ensure that sufficient power will be available to supply the safety-related equipment required for: (1) The safe shutdown of the facility; and (2) The mitigation and control of accident conditions within the facility. The minimum specified independent and redundant AC power and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50. The ACTIONS sections of the applicable Technical Specifications provide requirements specified for various levels of degradation of the power sources and provide restrictions upon continued facility operation commensurate with the level of degradation. The Operability of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite AC power sources and associated distribution systems operable during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite AC source.

The installation of the transformers with automatic load tap changers reduces the possibility of the loss of the offsite power system due to the increased grid voltage variations as documented in the description of the change in section 4.1.4. Therefore, the installation of the transformers with load tap changers will not 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:* John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* Stephen Dembek.

*Virginia Electric and Power Company, Docket No. 50–338, North Anna Power Station, Unit No. 1, Louisa County, Virginia*

*Date of amendment request:* January 9, 2001.

*Description of amendment request:* The proposed administrative changes will remove obsolete license conditions from the Facility Operating License (FOL) and implement associated changes to the Technical Specifications (TS). These changes involve editorial revisions, relocation of license conditions, removal of redundant license conditions covered throughout the license, removal of expired license conditions, and removal of license conditions and TS associated with completed modifications.

*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:

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

The proposed change to the North Anna Unit 1 Facility Operating License, NPF–4, is administrative (and in part editorial) in nature. The removal of license conditions regarding completed, no longer needed, and expired requirements has no impact on plant operations since these requirements no longer have meaningful applications. The renumbering and/or relocation within the FOL of various license conditions in this proposed administrative change does not alter the technical basis, requirements or the implementation of the affected items. The proposed change is within the current design and licensing bases of the facility. Since this change is administrative only and neither station operations nor design are affected by the change, it does not involve any significant increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

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

The proposed change is administrative (and in part editorial) in nature. The license conditions that are being removed or relocated by this proposed change do not impact station operations or station equipment in any manner. The proposed change does not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients that has not been previously analyzed. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. Consequently, no new failure modes are introduced and the proposed administrative change to the North Anna Unit 1 Facility Operating License does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

The proposed change is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. Since station operations are not affected by the proposed administrative change and no physical change is being made to the station, the change does not impact the condition, design, or performance of any station structure, system or component. Therefore, the proposed administrative change to the North Anna Unit 1 Facility Operating License does not involve a significant reduction in any margin of safety described in the bases of the Technical Specifications.

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:* Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

*NRC Section Chief:* Maitri Banerjee, Acting.

*Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit No. 2, Louisa County, Virginia*

*Date of amendment request:* January 9, 2001.

*Description of amendment request:* The proposed administrative changes will remove obsolete license conditions from the Facility Operating License (FOL). These changes involve editorial revisions, relocation of license conditions, removal of expired license conditions, and removal of license conditions associated with completed modifications.

*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:

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

The proposed change to the North Anna Unit 2 Facility Operating License, NPF–7, is administrative (and in part editorial) in nature. The removal of license conditions regarding completed, no longer needed, and expired requirements has no impact on plant operations since these requirements no longer have meaningful applications. The renumbering and/or relocation within the FOL of various license conditions in this proposed administrative change does not alter the technical basis, requirements or the implementation of the affected items. The proposed change is within the current design and licensing bases of the facility. Since this change is administrative only and neither station operations nor design are affected by the change, it does not involve any significant increase in the probability or the consequences of any accident or malfunction of equipment important to safety previously evaluated.

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

The proposed change is administrative (and in part editorial) in nature. The license conditions that are being removed or relocated by this proposed change do not impact station operations or station equipment in any manner. The proposed change does not involve a physical alteration of the plant, nor a change in the methods used to respond to plant transients that has not been previously analyzed. No new or different equipment is being installed and no installed equipment is being removed or operated in a different manner. Consequently, no new failure modes are introduced and the proposed administrative change to the North Anna Unit 2 Facility Operating License does not create the possibility of a new or different kind of accident or malfunction of equipment important to safety from any previously evaluated.

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

The proposed change is administrative (and in part editorial) in nature and neither station operations nor design are affected by the change. Since station operations are not affected by the proposed administrative change and no physical change is being made to the station, the change does not impact the condition, design, or performance of any station structure, system or component. Therefore, the proposed administrative change to the North Anna Unit 2 Facility Operating License does not involve a significant reduction in any margin of safety described in the bases of the Technical Specifications.

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:* Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

*NRC Section Chief:* Maitri Banerjee, Acting.

**Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing**

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

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

*PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey*

*Date of amendment request:* January 8, 2001.

*Brief description of amendment request:* The proposed amendment would revise the Technical Specifications (TS) to change the acceptance values for Core Spray subsystem flow contained in TS 4.5.1.b.1 from the current value of 6350 gallons per minute (gpm) to 6150 gpm.

*Date of publication of individual notice in* ***Federal Register:*** January 22, 2001 (66 FR 6701).

*Expiration date of individual notice:* February 21, 2001.

**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 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).

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

*Date of application for amendments:* June 16, 2000.

*Brief description of amendments:* The amendments revise TS Table 3.3.10–1, ''Post Accident Monitoring Instrumentation,'' to add the high pressure safety injection (HPSI) cold leg flow and HPSI hot leg flow instrumentation to the table.

*Date of issuance:* February 8, 2001.

*Effective date:* February 8, 2001, and shall be implemented within 30 days of the date of issuance.

*Amendment Nos.:* Unit 1—131 , Unit 2—131, Unit 3—131.

*Facility Operating License Nos. NPF–41, NPF–51, and NPF–74:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** October 4, 2000 (65 FR 59220)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 08, 2001.

No significant hazards consideration comments received: No.

*Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland*

*Date of application for amendments:* September 14, 2000.

*Brief description of amendments:* The amendments incorporate changes described below into the Technical Specifications for Culvert Cliffs Units 1 and 2. On September 9, 1996, a final rule amending 10 CFR 50.55a was issued requiring owners to implement, by September 9, 2001, the requirements of the 1992 Edition through the 1992 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Subsections IWE and IWL, as modified and supplemented by 10 CFR 50.55a. Calvert Cliffs Nuclear Power Plant, Inc. has developed a program to effect the implementation of Subsections IWE and IWL.

*Date of issuance:* January 30, 2001.

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

*Amendment Nos.:* 240 and 214.

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

*Date of initial notice in* ***Federal Register:*** October 18, 2000 (65 FR 62384).

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated January 30, 2001.

No significant hazards consideration comments received: No.

*Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan*

*Date of application for amendment:* November 19, 1999, as supplemented May 31, August 2, October 19, and November 21, 2000.

*Brief description of amendment:* The amendment revises the Technical Specifications (TSs) by changing (1) the design features description of the fuel storage equipment and configuration to allow an increase in the spent fuel pool (SFP) storage capacity and (2) the description of the high-density spent fuel racks program to clarify that the surveillance program is applicable only to racks containing Boraflex as a neutron absorber. Specifically, the amendment revises the TSs for Fermi 2 to increase the capacity of the SFP from 2,414 to 4,608 fuel assemblies.

*Date of issuance:* January 25, 2001.

*Effective date:* As of the date of issuance and shall be implemented within 90 days.

*Amendment No.:* 141.

*Facility Operating License No. NPF–43:* Amendment revises the Technical Specifications

*Date of initial in **Federal Register*** March 13, 2000 (65 FR 13336)

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

No significant hazards consideration comments received: No.

*Detroit Edison Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan*

*Date of application for amendment*: September 20, 2000.

*Brief description of amendment:* The amendment changes Technical Specification (TS) 5.5.7.d to decrease the maximum allowed pressure drops across control room emergency filtration (CREF) make-up and recirculation train filters and charcoal absorbers. The words ''(CREF only)'' are also removed from the TS.

*Date of issuance:* February 8, 2001.

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

*Amendment No.*: 142.

*Facility Operating License No. NPF–43:* Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 1, 2000 (65 FR 65340).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 8, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* November 30, 2000.

*Brief description of amendment:* The amendment relocated the boration systems requirements from the Technical Specifications to the Technical Requirements Manual.

*Date of issuance:* January 31, 2001.

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

*Amendment No.:* 229.

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

*Date of initial notice in **Federal Register:*** December 27, 2000 (65 FR 81916).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 2001.

No significant hazards consideration comments received: No.

*Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York*

*Date of application for amendment:* November 19, 1999, as supplemented October 12, 2000.

*Brief description of amendment:* The amendment changes the Technical Specification surveillance testing requirements of the charcoal adsorbers in the Standby Gas Treatment System and the Control Room Emergency Ventilation Air Supply System to meet the requested actions of Generic Letter 99–02.

*Date of issuance:* February 5, 2001.

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

*Amendment No.:* 269.

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

*Date of initial notice in **Federal Register:*** February 9, 2000 (65 FR 6410).

The October 12, 2000, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* May 22, 2000, as supplemented October 4, 2000.

*Brief description of amendments:* Changed the Technical Specifications to incorporate that portion of the August 8, 1996, Final Amended Rule (61 FR 41303) related to revised requirement of inservice inspection of the containment post-tensioning system.

*Date of issuance:* January 31, 2001.

*Effective date:* January 31, 2001.

*Amendment Nos.* 210 and 204.

*Facility Operating License Nos. DPR–31 and DPR–41:* Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48750). The October 4, 2000 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* December 6, 2000.

*Brief description of amendments:* The amendments delete Technical Specifications (TS) Section 6.8.4.d, ''Post Accident Sampling,'' for Turkey Point Units 3 and 4 and thereby eliminate the requirements to have and maintain the post-accident sampling system (PASS) for those units.

*Date of issuance:* January 31, 2001.

*Effective date:* January 31, 2001.

*Amendment Nos.* 211 and 205.

*Facility Operating License Nos. DPR–31 and DPR–41:* Amendments revised the TSs.

*Date of initial notice in **Federal Register:*** December 27, 2000 (65 FR 81923).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 2001.

No significant hazards consideration comments received: No.

*Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan*

*Date of application for amendment:* September 30, 2000, as supplemented November 22, and December 20, 2000.

*Brief description of amendment:* The amendment would allow an extension of the steam generator tube inspection surveillance requirements of Technical Specification Surveillance Requirement 4.4.5.3. Specifically, the licensee requested to extend the required inspection interval from 40 to 56 calendar months.

*Date of issuance:* January 30, 2001.

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

*Amendment No.:* 232.

*Facility Operating License No. DPR–74:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** October 18, 2000 ( 65 FR 62387).

The supplemental information contained clarifying information and 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 January 30, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* June 1, 2000.

*Brief description of amendment:* The amendment approves changes to Technical Specifications (TSs) 3.3.3.2, ''Instrumentation, Movable Incore Detectors''; 3.3.3.3, ''Instrumentation, Seismic Instrumentation''; 3.3.3.4, ''Instrumentation, Meteorological Instrumentation''; 3.3.3.8, ''Loose-Part Detection System''; 3.3.4, ''Turbine Overspeed Protection''; and Index Pages vi and vii. The changes relocate the requirements for the incore detectors, seismic instrumentation, meteorological instrumentation, loose-part detection system, and turbine overspeed protection system from the TSs to the Technical Requirements Manual. The Bases for these TSs have been modified to reflect the TS changes.

*Date of issuance:* January 29, 2001.

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

*Amendment No.:* 193.

*Facility Operating License No. NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 29, 2000 (65 FR 71136).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* July 31, 2000 as supplemented January 5, 2001.

*Brief description of amendment:* The amendment changes Technical Specifications (TSs) 3.8.1.1, ''Electrical Power Systems—A.C. Sources—Operating,'' and 3.8.1.2, ''Electrical Power Systems—A.C. Sources—Shutdown.'' The changes allow certain EDG surveillance requirements to be performed when the plant is operating instead of shut down as currently required. The index and Bases for these TSs are modified to reflect the changes.

*Date of issuance:* February 2, 2001.

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

*Amendment No.:* 194.

*Facility Operating License No. NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** September 6, 2000 (65 FR 54087).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 2, 2001.

No significant hazards consideration comments received: No.

*Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin*

*Date of application for amendment:* November 18, 1999, as supplemented August 7, 2000.

*Brief description of amendment:* The amendment to the Kewaunee Nuclear Power Plant Technical Specifications approves an increase in the allowable number of spent fuel assemblies in the spent fuel pools. The addition of the 215 storage locations in the new north canal pool will extend the full-core reserve capability until after the 2009 outage, and increase the total capacity to 1,205 spent fuel assemblies.

*Date of issuance:* January 23, 2001.

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

*Amendment No.:* 150.

*Facility Operating License No. DPR–43:* Amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 1 and December 21, 2000 (65 FR 65347 and 65 FR 80471 respectively)

The supplemental information contained clarifying information and 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 January 23, 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 12, 2000, as supplemented by letter dated January 25, 2001.

*Brief description of amendments:* These amendments authorize (1) a design upgrade of the refueling water purification (RWP) system to permit reclassification of this system from Design Class II/non-Seismic Category 1 to Design Class I/Seismic Category 1 to allow filtering of the refueling water storage tank (RWST) water while the RWST is required to be operable, and (2) the use of a temporary reverse osmosis skid mounted system to reduce RWST silica concentration levels while the RWST is required to be operable following upgrade of the RWP system to satisfy reactor coolant chemistry limits.

*Date of issuance:* January 29, 2001.

*Effective date:* January 29, 2001, and shall be implemented in the next periodic update to the FSAR Update, following upgrade of the refueling water purification system, in accordance with 10 CFR 50.71(e).

*Amendment Nos.:* Unit 1—144 ; Unit 2–143.

*Facility Operating License Nos. DPR–80 and DPR–82:* The amendments revised the FSAR Update.

*Date of initial notice in* ***Federal Register:*** July 12, 2000 (65 FR 43050).

The January 25, 2001, supplemental letter 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 amendments is contained in a Safety Evaluation dated January 29, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* September 6, 2000 (PCN–274, Supplement 1).

*Brief description of amendments:* The amendments revised the San Onofre, Units 2 and 3 Technical Specification 3.3.11, ''Post Accident Monitoring Instrumentation (PAMI),'' to extend the PAMI surveillance frequency from 18 to 24 months to accommodate a 24-month fuel cycle.

*Date of issuance:* January 30, 2001.

*Effective date:* January 30, 2001, to be implemented within 30 days of issuance.

*Amendment Nos.:* Unit 2—176; Unit 3–167.

*Facility Operating License Nos. NPF–10 and NPF–15:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** October 18, 2000 (65 FR 62391).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 30, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* October 6, 2000 (PCN–518).

*Brief description of amendments:* The amendments revise TS 3.7.11, ''Control Room Emergency Air Cleanup System (CREACUS),'' to establish actions to be taken for inoperable ventilation systems due to a degraded control room pressure boundary. The amendments allow up to 24 hours to restore the pressure boundary to operable status when two ventilation trains are inoperable due to an inoperable pressure boundary in Modes 1, 2, 3, and 4. In addition, a limiting condition for operation note is added to allow the pressure boundary to be opened intermittently under administrative control without affecting CREACUS operability.

*Date of issuance:* January 30, 2001.

*Effective date:* January 30, 2001, to be implemented within 30 days of issuance.

*Amendment Nos.:* Unit 2—177; Unit 3—168.

*Facility Operating License Nos. NPF–10 and NPF–15:* The amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 15, 2000 (65 FR 69066).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 30, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* November 10, 2000.

*Brief description of amendment:* This amendment will allow: (a) the minimum fuel oil stored in the fuel oil storage tank (FOST) for each emergency diesel generator (EDG) to be raised from 47,100 gallons to 48,500 gallons for Modes 1–4, and from 33,200 gallons to 42,500 gallons for Modes 5 and 6; and (b) the minimum fuel oil maintained in the day fuel tank for each EDG to be raised from 300 gallons to 360 gallons for Modes 1–6.

*Date of issuance:* February 2, 2001.

*Effective date:* February 2, 2001.

*Amendment No.:* 150.

*Facility Operating License No. NPF–12:* Amendment revises the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** November 20, 2000 (65 FR 69795).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 2, 2001.

No significant hazards consideration comments received: No.

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

*Date of amendments request:* October 9, 2000, supplemented December 4, 2000.

*Brief Description of amendments:* The amendments revise Technical Specification 5.5.14, ''Technical Specification (TS) Bases Control Program,'' to provide consistency with the changes in 10 CFR 50.59 which were published in the ***Federal Register*** on October 4, 2000.

*Date of issuance:* January 31, 2001.

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

*Amendment Nos.:* 148 and 140.

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

*Date of initial notice in* ***Federal Register:*** December 20, 2000 (65 FR 79907) The supplement dated December 4, 2000, provided clarifying information that did not change the scope of the October 4, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 2001.

No significant hazards consideration comments received: No.

*Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia*

*Date of application for amendments:* June 14, 2000.

*Brief description of amendments:* The amendments revised the Technical Specifications (TSs) Surveillance Requirements (SR) 3.8.1.9 and 3.8.1.14 to reduce diesel generators loading requirements from ≥ 6800 kW and ≤ 7000 kW to ≥ 6500 kW and ≤ 7000 kW. These changes will make the above SRs consistent with SRs 3.8.1.3 and 3.8.1.13, which are in the current TSs. In addition, the proposed changes would correct a typographical error in Section 5.6.7, ''EDG Failure Report,'' in the Vogtle TS. This editorial change will correctly reference Regulatory Position C.4 of Regulatory Guide 1.9, Revision 3 instead of Regulatory Position C.5.

*Date of issuance:* January 31, 2001.

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

*Amendment Nos.:* Unit 1—117; Unit 2—95.

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

*Date of initial notice in* ***Federal Register:*** September 6, 2000 (65 FR 54087).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 2001.

No significant hazards consideration comments received: No.

*Tennessee Valley Authority, Docket Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 2 and 3, Limestone County, Alabama*

*Date of application for amendments:* August 11, 2000 (TS–400) as supplemented by letter dated October 20, 2000.

*Brief description of amendments:* The amendments revised the surveillance test requirements for excess flow check valves.

*Date of issuance:* January 29, 2001.

*Effective date:* January 29, 2001.

*Amendment Nos.:* 268 and 228.

*Facility Operating License Nos. DPR–52 and DPR–68:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** September 6, 2000 (65 FR 54088). The October 20, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 2001.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* December 7, 2000 (ET 00–0041).

*Brief description of amendment:* The amendment changes Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' of the Technical Specifications. The change adds Surveillance Requirement (SR) 3.3.2.10 for the following two engineered safety feature actuation system instrumentation in the table: item 6.f, loss of offsite power, and item 6.h, auxiliary feedwater pump suction transfer on suction pressure—low.

*Date of issuance:* February 06, 2001.

*Effective date:* February 06, 2001, and shall be implemented including the changes to the Bases for the response times, within 60 days of the date of issuance.

*Amendment No.:* 136.

*Facility Operating License No. NPF–42.* The amendment revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** December 27, 2000 (65 FR 81932).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 06, 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 20852, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By March 23, 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).

*Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York*

*Date of amendment request:* December 19, 2000.

*Description of amendment request:* The amendment revises the Technical Specifications to indicate that quadrant power tilt limits apply only when reactor power is greater than 50 percent.

*Date of issuance:* December 20, 2000.

*Effective Date:* As of its date of issuance and shall be implemented within 30 days.

*Amendment No.:* 204.

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

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

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 December 20, 2000.

*Attorney for licensee:* Mr. John M. Fulton, Assistant General Counsel Entergy Nuclear Generating Co. Pilgrim Station, 600 Rocky Hill Road Plymouth, MA 02360.

*NRC Section Chief:* Marsha Gamberoni.

Dated at Rockville, Maryland, this 14th day of February 2001.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 01–4228 Filed 2–20–01; 8:45 am]

**BILLING CODE 7590–01–U**

## SECURITIES AND EXCHANGE COMMISSION

## Proposed Collection; Comment Request

*Upon Written Request, Copies Available From:* Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

*Extension:*

Form SE, OMB Control No. 3235–0327, SEC File No. 270–289;
Form ID, OMB Control No. 3235–0328, SEC File No. 270–291;
Form ET, OMB Control No. 3235–0329, SEC File No. 270–290; and
Form TH, OMB Control No. 3235–0425, SEC File No. 270–377.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (''Commission'') is soliciting comments on the collections of information summarized below. The Commission plans to submit these existing collections of information to the Office of Management and Budget for extension and approval.