www.nrc.gov/reading-rm/adams/html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, or 301–415–4737 or by email to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 29th day of December, 2003.

For the Nuclear Regulatory Commission.

Travis Tate,

Project Manager, Section 2, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 04–186 Filed 1–5–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATES: Weeks of January 5, 12, 19, 26, February 2, 9, 2004.

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland.

STATUS: Public and closed.

MATTERS TO BE CONSIDERED:

Week of January 5, 2004

There are no meetings scheduled for the Week of January 5, 2004.

Week of January 12, 2004—Tentative

Wednesday, January 14, 2004

9:30 a.m. Briefing on Status of Office of Chief Information Officer Programs, Performance, and Plans (Public Meeting) (Contact: Jacqueline Silber, 301–415–7330).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

Week of January 19, 2004—Tentative

Wednesday, January 21, 2004

1:30 p.m. Discussion of Security Issues (Closed—Ex. 1).

Week of January 26, 2004—Tentative

There are no meetings scheduled for the Week of January 26, 2004.

Week of February 2, 2004—Tentative

There are no meetings scheduled for the Week of February 2, 2004.

Week of February 9, 2004—Tentative

There are no meetings scheduled for the Week of February 9, 2004.

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Timothy J. Frye, (301) 415–1651.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

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, DC 20555, (301) 415–1969. In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: December 31, 2003.

R. Michelle Schroll,

Information Management Specialist, Office of the Secretary.

[FR Doc. 04–311 Filed 1–2–04; 12:08 pm]

BILLING CODE 7590–01–M

UNITED STATES 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 December 12, 2003, through December 23, 2003. The last biweekly notice was published on December 23, 2003 (68 FR 74262).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By February 5, 2004, 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendment request: December 2, 2003.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) Surveillance Requirement (SR) 4.0.2 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is less* * *'' to ''* * *up to 24 hours or up to the limit of the specified frequency, whichever is greater.* * *'' To support this change, the following requirement would be added to SR 4.0.2: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' Additionally, a new section 6.2.1 will be added to provide for a TS Bases Control Program.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, 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 September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application. The NSHC determination is restated below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in [a] margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on [a] margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

Therefore, this change does not involve a significant reduction in a 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.

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

NRC Section Chief: Richard J. Laufer.

Carolina Power & Light Company, Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: October 31, 2003.

Description of amendment request: The proposed amendment would revise the Minimum Critical Power Ratio (MCPR) Safety Limit contained in Technical Specification (TS) 2.1.1.2. Currently the MCPR value is greater than or equal to 1.12 for two recirculation loop operation and greater than or equal to 1.14 for single recirculation loop operation. The proposed revised MCPR would be greater than or equal to 1.11 for two recirculation loop operation and greater than or equal to 1.12 for single recirculation loop operation. Also, a second proposed change would add topical report NEDE–32906P–A, ''TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses,'' to the list of documents specified in TS 5.6.5 describing the approved methodologies used to determine the core operating limits.

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

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

Response: No.

Proposed Change 1

The proposed change to Technical Specification 2.1.1.2 does not alter the assumptions of the accident analyses or the Technical Specification Bases. The MCPR Safety Limit values are calculated to ensure that greater than 99.9 percent of the fuel rods in the core avoid transition boiling during any plant operation if the safety limit is not violated. The derivation of the MCPR Safety Limit values specified in the Technical Specifications has been performed using the methods discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (*i.e.,* GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June 2000, which incorporates Amendment 26. By letters dated November 10, 1999, and March 29, 2000, GNF, the NRC approved the use of Amendment 26 to NEDE–24011–P–A. Appropriate operational MCPR limits are applied that ensure the MCPR Safety Limit is not exceeded during all modes of operation and anticipated operational occurrences.

The revised MCPR Safety Limit values do not affect the operability of any plant systems nor do these revised values compromise any fuel performance limits; therefore, the probability of fuel damage will not be increased as a result of this change. The MCPR Safety Limit values do not impact the source term or pathways assumed in accidents previously evaluated, and there are no adverse effects on the factors contributing to offsite or onsite radiological doses. In addition, the revised MCPR Safety Limit values do not affect the performance of any equipment used to mitigate the consequences of a previously evaluated accident and do not affect setpoints that initiate protective or mitigative actions.

Therefore, the proposed change to MCPR Safety Limit values contained in Technical Specification 2.1.1.2 does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change 2

The proposed change to TS 5.6.5 will add General Electric Nuclear Energy topical report NEDE–32906P–A, ''TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses,'' to the list of documents describing approved methodologies for determining core operating limits. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits, which are developed using the topical report being added, ensure that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed change does not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. Use of the analytical methodologies described in the topical report being added to TS 5.6.5 will ensure that applicable design and safety analyses acceptance criteria are met. Use of these NRC-approved methodologies does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. Use of the approved methodologies described in the topical report being added to TS 5.6.5 ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

Therefore, the proposed change adding General Electric Nuclear Energy topical report NEDE–32906P–A to the TS 5.6.5 list of documents describing approved methodologies for determining core operating limits does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Proposed Change 1

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. The proposed revision of the MCPR Safety Limit values does not involve installation of any new or different equipment. No installed equipment is being operated in a different manner than currently evaluated. No new initiating events or transients will result from use of the revised MCPR Safety Limit values. As a result, no new failure modes are being introduced. Therefore, the proposed change to MCPR Safety Limit values contained in Technical Specification 2.1.1.2 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change 2

The proposed change adding topical report NEDE–32906P–A to TS 5.6.5, and the use of the analytical methods described therein, does not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel and core designs in accordance with NRC approved methodologies. The proposed methodology continues to meet applicable criteria for core operating limit analysis. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced.

Therefore, the proposed change adding General Electric Nuclear Energy topical report NEDE–32906P–A to the TS 5.6.5 list of documents describing approved methodologies for determining core operating limits does 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?

Response: No.

Proposed Change 1

The margin of safety is established through the design of the plant structures, systems, and components; through the parameters within which the plant is operated; through the establishment of setpoints for actuation of equipment relied upon to respond to an event; and through margins contained within the safety analyses. The revised MCPR Safety Limit values will not adversely impact the performance of plant structures, systems, components, and setpoints relied upon to respond to mitigate an accident or transient. The MCPR Safety Limit values are calculated to ensure that greater than 99.9 percent of the fuel rods in the core avoid transition boiling during any anticipated operation occurrences if the safety limit is not violated, thereby ensuring that fuel cladding integrity is maintained. The revised MCPR Safety Limit values have been calculated using NRC approved methods and procedures and preserve the existing margin to transition boiling. Based on the assurance that the fuel design criteria are being met, the revised MCPR Safety Limit values do not involve a reduction in a margin of safety.

Proposed Change 2

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change adding General Electric Nuclear Energy topical report NEDE–32906P–A to the TS 5.6.5 list of documents describing approved methodologies for determining core operating limits does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change adding topical report NEDE–32906P–A to the TS 5.6.5 list of documents describing approved methodologies for determining core operating limits does not result in a significant reduction in the margin of safety.

Based on the above, PEC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen G. Howe.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois, and Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: November 3, 2003.

Description of amendment request: The proposed amendment would modify Technical Specifications (TS) 3.4.1, ''Recirculation Loops Operating,'' to add a requirement for the linear heat generation rate (LHGR) limits specified in the Core Operating Limits Report (COLR) to be met during single recirculation loop operation.

Technical Specification 3.4.1 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, LaSalle County Station

(LSCS) Units 1 and 2, and Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2, currently requires limits for average planar linear heat generation rate (APLHGR) and minimum critical power ratio (MCPR), as well as allowable values for certain Reactor Protection System and Control Rod Block functions, to be modified during single recirculation loop operation. The modified limits for APLHGR and MCPR are specified in the COLR. The proposed change adds a requirement to modify the LHGR limit as specified in the COLR with one recirculation loop in operation. Although there is currently no TS requirement to adjust the LHGR limit during single recirculation loop operation, in accordance with NRC Administrative Letter 98–10, ''Dispositioning of Technical Specifications that Are Insufficient to Assure Plant Safety,'' administrative controls are in place at DNPS and QCNPS to ensure that the LHGR limits are appropriately adjusted.

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 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. The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences, and to ensure that the peak cladding temperature (PCT) during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46.

LHGR limits have been established consistent with the NRC-approved GESTAR methodology to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change establishes a requirement for LHGR limits to be modified, as specified in the COLR, during SLO such that the fuel is protected during SLO and during any plant transients or anticipated operational occurrences that may occur while in SLO.

Modifying the LHGR limits during SLO does not increase the probability of an evaluated accident. The proposed change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences, and to ensure that the PCT during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. This will ensure that the fuel design safety criteria (*i.e.*, less than 1% plastic strain of the fuel cladding and no fuel centerline melting) are met and that the core remains in a coolable geometry following a postulated design basis LOCA. Since the operability of plant systems designed to mitigate any consequences of accidents has not changed, the consequences of an accident previously evaluated are not expected to increase.

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.

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. The proposed change does not involve any modifications of the plant configuration or allowable modes of operation. Requiring the LHGR limits to be modified for SLO by applying the SLO LHGR multiplier ensures that the assumptions of the LOCA analyses are met.

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

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change will not adversely affect operation of plant equipment. The change will not result in a change to the setpoints at which protective actions are initiated. LHGR limits during SLO are established to ensure that the PCT during a postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. This will ensure that the core remains in a coolable geometry following a postulated design basis LOCA. The proposed change will ensure the appropriate level of fuel protection.

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 requested amendments involve no significant hazards consideration.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 16, 2003.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 3/4.4.5, ''Reactor Coolant System—Steam Generators,'' to allow a one-time extension of the steam generator tube inservice inspection interval from March 2004 to March 2005.

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

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

Response: No.

The steam generator tubes perform both an accident prevention and an accident mitigation function. Steam generator tube integrity is necessary to prevent the loss of reactor coolant system inventory to the secondary system and to provide a barrier to fission product release to the environment. The layup and storage conditions of the steam generator during the extended outage have been assessed and determined to not adversely affect steam generator conditions. An operational assessment of the steam generators for approximately 1.4 effective full power year has been performed to assure acceptable structural integrity during the extended surveillance interval. The operational assessment for the steam generators has determined that primary-to-secondary leakage following a steam line break, which is the limiting event (other than a tube rupture), would continue to be acceptable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change does not introduce any new or different failure mechanism for the steam generators. Steam generator tube integrity will be maintained as previously analyzed following postulated events. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The layup and storage conditions of the steam generator during the extended outage have been assessed and determined to not adversely affect steam generator condition. The operational assessment for the mid-cycle outage has shown that structural margins are greater at approximately 1.4 EFPY then they would be at the end of a typical full cycle of operation. Accident induced leakage is projected to be the same for the surveillance interval extension period as it would be for a full cycle of operation. 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: January 30, 2003.

Description of amendment request: This license amendment request proposes a revision to the reactor pressure vessel (RPV) material surveillance program described within the Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR) from a plant-specific program to the Boiling-Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP).

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.

NRC [Nuclear Regulatory Commission] regulations impose requirements upon the reactor coolant system to ensure that adequate safety margins against nonductile or rapidly propagating failures exits during normal operation, anticipated operational occurrences, and system hydrostatic tests. These requirements are set forth in 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants,'' Criterion 31, ''Fracture Prevention of Reactor Coolant Pressure Boundary,'' Appendix G, ''Fracture Toughness Requirements,'' and Appendix H requires that changes in the fracture toughness properties of reactor vessel materials, resulting from the neutron irradiation and the thermal environment, are monitored by a material surveillance program. To determine the effects of neutron fluence on the nil-ductility reference temperature of reactor vessel materials, the methods provided in Regulatory Guide (RG) 1.99, ''Radiation Embrittlement of Reactor Vessel Materials,'' Revision 2 are used.

As described in the PNPP USAR, the current PNPP material surveillance program is a plant-specific program which complies with 10 CFR 50, Appendix H.

The proposed amendment involves changing the material surveillance program from a plant-specific program to an integrated surveillance program. The use of an integrated program is consistent with the requirements of 10 CFR 50, Appendix H, Paragraph III.C. The integrated program proposed by PNPP is the BWRVIP ISP. The BWRVIP ISP has been reviewed and approved by the NRC staff as an acceptable program and is in conformance with 10 CFR 50, Appendix H. Use of the ISP, among its many benefits, will increase the number of data points used in the evaluation of changes in vessel material properties. This will improve compliance with the aforementioned NRC regulations. The methods contained in RG 1.99, Revision 2, will still be used to determine the effects of neutron fluence upon the nil-ductility reference temperature of the PNPP reactor vessel materials.

This change will not affect the reactor pressure vessel, as no physical changes are involved. The proposed change will not cause the reactor pressure vessel or interfacing systems to be operated outside of any design or testing limits. Furthermore, the proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of any accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change revises the PNPP licensing bases to reflect participation in the BWRVIP ISP. The ISP was approved by the NRC staff as an acceptable material surveillance program which complies with 10 CFR 50, Appendix H. The proposed change will not impact the manner in which the plant is designed or operated. No new accident types or failure modes will be introduced as a result of the proposed change. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from that previously evaluated.

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

The material surveillance program requirements contained in 10 CFR 50, Appendix H, provide assurance that adequate margins of safety exist for the reactor coolant system against nonductile or rapidly propagating failures during normal operation, anticipated operational occurrences, and safety hydrostatic tests. The BWRVIP ISP has been approved by the NRC staff as an acceptable material surveillance program which complies with 10 CFR 50, Appendix H. The ISP will provide the material surveillance data which will ensure that the safety margins require by NRC regulations are maintained for the PNPP reactor coolant system. Therefore, the proposed change does not involve a significant reduction in any margins of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: August 14, 2003.

Description of amendment request: This license amendment request (LAR) proposes a revision to increase the analytical limit and the resulting Technical Specification (TS) allowable value (AV) related to the setpoint for the Main Steam Line Turbine Building Temperature—High, system isolation function. This LAR revises the main steam line trip setpoint AV based on improved computer modeling of the expected building temperature transients in the event of a larger steam leak. The proposed change improves the operating margins and reduces challenges to the plant by avoiding unnecessary plant shutdown transients from turbine building high temperatures from other than a main steam line leak.

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 nuclear boiler Leak Detection System (LDS) instrumentation associated with the proposed amendment assists in the detection of a small steam leak to prevent a significant release of radioactive material created by conditions other than a break within the Reactor Coolant Pressure Boundary (RCPB).

The proposed amendment establishes a new steam leak system isolation temperature limit in the Turbine Building.

There is no accident analysis or transient that credits the subject LDS instrumentation. The subject instrumentation is for the detection of small steam leaks and not a pipeline break as described in the Updated Safety Analysis Report (USAR) Chapter 15 accident analysis. The detection of main steam line flow is the parameter used in the accident analysis to signal a steam line break outside of containment.

The proposed amendment does not impact the physical design or location of the LDS instrumentation. This proposed amendment is associated only with the results of a main steam line leak in the non-safety related Turbine Building and has no impact on the initiation of this leak. The analysis completed in support of the proposed amendment indicates that the radiological effects associated with the new steam leak system isolation limit remains bounded by the existing large main steam line break analysis contained within the PNPP [Perry Nuclear Power Plant] USAR. The proposed leakage limit does not alter the current function of the LDS that isolates the Main Steam system prior to the leakage degrading to a point where the system integrity, *i.e.*, piping integrity and makeup capability, is challenged. Therefore, the proposed amendment ensures that the criteria for acceptance as established in the original licensing bases and the requirements of the original design basis remain valid. It has been determined that the service life, *i.e.*, Equipment Qualification (EQ) and structural integrity of the Structures, Systems and Components (SSC) in the affected areas are not adversely impacted by the proposed amendment.

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

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

The proposed amendment does not impact the physical design or location of the associated LDS instrumentation. The instruments will still promptly initiate the automatic isolation of the appropriate Containment and Drywell isolation valves to mitigate steam leakage as credited in the original licensing bases. This proposed amendment is associated only with the results of a main steam line break in the non-safety related Turbine Building and has no impact on the initiation of this leak. The analysis completed in support of the proposed amendment indicates that the radiological effects associated with the new steam leak system isolation limit remains bounded by the existing large main steam line break analysis contained within the PNPP USAR. The EQ and structural integrity of any SSC located within the non-safety related Turbine Building are not affected by the proposed amendment. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The analysis performed for the proposed amendment proves that the appropriate instruments will still promptly initiate automatic system isolation, upon sensing temperatures in excess of their setpoints. The radiological effects associated with the proposed small steam leak to be detected remain bounded by the existing large main steam line break analysis contained within the USAR. Steam leaks in the affected area of the Turbine Building will be detected on a timely basis so that the Main Steam system will be isolated before such degradation could become sufficiently severe to jeopardize the safety of the system. Also, steam leaks will be detected before the leakage could increase to a level beyond the capability of the makeup system. Therefore, the proposed amendment ensures that the criteria for acceptance as established in the original licensing bases and the requirements of the original design basis remain valid. There is no accident analysis or transient that credits the associated leak detection instrumentation, and the LDS Main Steam Line Turbine Building Temperature—High function is categorized as non-risk significant. Further, the proposed amendment reduces the challenges to SSCs due to unnecessary plant shutdowns created by conditions other than a main steam line leak. The EQ and structural integrity of any SSC located within the Turbine Building are not affected by the proposed amendment. Therefore, the proposed amendment 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 29, 2003.

Description of amendment request: The proposed license amendments would allow relocation of specific pressure and flow values for the boric acid makeup (BAM) pumps, containment spray (CS) pumps, high pressure safety injection (HPSI) pumps, and low pressure safety injection (LPSI) pumps from the St. Lucie Units 1 and 2 Technical Specifications to the Updated Final Safety Analysis Reports (UFSARs). This is consistent with the Combustion Engineering Improved Standard Technical Specifications and the Nuclear Regulatory Commission Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors.

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

(1) Would operation of the facility in accordance with the proposed amendments involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes to relocate the BAM, CS, HPSI, and LPSI pump surveillance verification details in the aforementioned Technical Specifications surveillance requirements to the St. Lucie UFSARs do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility, or the manner in which it is operated. The proposed changes do not alter or prevent the ability of structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the St. Lucie UFSARs.

The subject surveillance requirement criteria relocated to the St. Lucie UFSARs will continue to be administratively controlled. Changes to the St. Lucie UFSARs are evaluated and controlled under 10 CFR 50.59 prior to implementation. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Would operation of the facility in accordance with the proposed amendments create the possibility of a new different kind of accident from any accident previously evaluated?

The proposed changes do not alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated.

There are no changes to the source term or radiological release assumptions used in evaluating the radiological consequences in the St. Lucie UFSARs. The proposed changes have no adverse impact on the component or system interactions. The proposed changes will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the UFSARs. The proposed changes do not change the level of programmatic and procedural details of assuring operation of the facility in a safe manner. Since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new different kind of accident from any previously analyzed.

(3) Would operation of the facility in accordance with the proposed amendments involve a significant reduction in a margin of safety?

There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes do not reduce the level of programmatic or procedural controls associated with the activities presently performed via the aforementioned surveillance requirements.

Future changes to the relocated technical requirements will require an evaluation pursuant to the provisions of 10 CFR 50.59 prior to implementation.

Therefore, relocation of the specific pump pressure and flow criteria contained in the aforementioned Technical Specification Surveillance Requirements to the St. Lucie Units 1 and 2 UFSARs does not involve a significant reduction in the margin of safety provided in the existing 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: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: November 21, 2003.

Description of amendment request: The proposed amendments would transfer Technical Specification (TS) requirements 6.5 (Review and Audit), 6.8.2 and 6.8.3 (procedures and programs review specifics), and 6.10 (Record Retention) to the quality assurance plan (a licensee controlled document) for St. Lucie Units 1 and 2.

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

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

The proposed changes to the St. Lucie Plant TS do not adversely affect accident initiators or precursors, nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. In addition, the proposed changes do not affect the manner in which the plant responds in normal operation, transient, or accident conditions, nor do they change any of the procedures related to operation of the plant. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed changes are administrative for the purpose of updating TS to reflect current NRC and industry initiatives.

The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the St. Lucie UFSARs. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released off site, nor significantly increase individual or cumulative occupational/public radiation exposures.

Therefore, it is concluded that these proposed revisions do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

The proposed changes to the St. Lucie Plant TS do not change the operation or the design basis of any plant system or component during normal or accident conditions. The proposed changes do not include any physical changes to the plant. In addition, the proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes.

The proposed changes are administrative in nature and only correct, update and clarify the St. Lucie Plant Technical Specifications to reflect NRC guidance, *i.e.*, AL 95–06. The proposed changes do not modify the facility nor do they affect the plant's response to normal, transient, or accident conditions. The changes do not introduce a new mode of plant operation. The changes are an enhancement and do not affect plant safety. The plant's design and design basis are not revised and the current safety analyses remains in effect.

Thus, these proposed revisions to the St Lucie Plant TS do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes are administrative changes to the St. Lucie Plant Technical Specifications. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications are not revised nor is the plant design or its method of operation revised by the proposed changes.

Thus, it is concluded that these proposed revisions to the St. Lucie Plant TS do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: December 6, 2003.

Description of amendment request: The proposed amendment would revise the Unit 1 and 2 Technical Specifications (TSs) by adding a requirement to apply linear heat generation rate [LHGR] limits if the main turbine bypass system becomes inoperable.

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

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

Response: No.

The proposed change to the TS 3.7.6 does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the MCPR [minimum critical power ratio] and LHGR limits documented in the unit/cycle specific COLRs [core operating limits report] for Main Turbine Bypass System operable and inoperable are generated using NRC [Nuclear Regulatory Commission] approved methodology and meet the applicable acceptance criteria. The COLR operating limits thus assure that the MCPR Safety Limit and LHGR Limit will not be exceeded during normal operation or anticipated operational occurrences. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence or 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?

Response: No.

The proposed change to TS 3.7.6 does not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, the proposed changes do not create the possibility of a previously unevaluated operator error or a new single failure.

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

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

Response: No.

Since the proposed changes do not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed change to TS 3.7.6 does not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR and LHGR limits calculated for Main Turbine Bypass System operable and inoperable preserve the required margin of safety.

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

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc, General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101,1179.

NRC Section Chief: Richard J. Laufer.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: October 22, 2003 (TSC 03–12).

Description of amendment request: The proposed change involves the extension from 1 hour to 24 hours of the completion time (CT) for Action (a) of Technical Specification (TS) 3.5.1.1, which defines requirements for accumulators. Accumulators are part of the emergency core cooling system and consist of tanks partially filled with borated water and pressurized with nitrogen gas. The contents of the tank are discharged to the reactor coolant system (RCS) if, as during a loss-of-coolant accident, the coolant pressure decreases to below the accumulator pressure. Action (a) of TS 3.5.1.1 specifies a CT to restore an accumulator to operable status when it has been declared inoperable for a reason other than the boron concentration of the water in the accumulator not being within the required range. This change was proposed by the Westinghouse Owners Group participants in the TS Task Force (TSTF) and is designated TSTF–370. TSTF–370 is supported by NRC-approved topical report WCAP–15049–A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' submitted on May 18, 1999. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on July 15, 2002 (67 FR 46542), on possible amendments concerning TSTF–370, 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 March 12, 2003 (68 FR 11880). The licensee affirmed the applicability of the following NSHC determination in its application dated October 22, 2003.

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 basis for the accumulator limiting condition for operation (LCO), as discussed in Bases Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of the increase in the accumulator CT on core damage frequency for all the cases evaluated in WCAP–15049–A is within the acceptance limit of 1.0E–06/yr for a total plant core damage frequency (CDF) less than 1.0E–03/yr. The incremental conditional core damage probabilities calculated in WCAP–15049–A for the accumulator CT increase meet the criterion of 5E–07 in Regulatory Guides (RG) 1.174, ''An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,'' and 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for all cases except those that are based on design basis success criteria. As indicated in WCAP–15049–A, design basis accumulator success criteria are not considered necessary to mitigate large break loss-of-coolant accident (LOCA) events, and were only included in the WCAP–15049–A evaluation as a worst case data point. In addition, WCAP–15049–A states that the NRC has indicated that an incremental conditional core damage frequency (ICCDP) greater than 5E–07 does not necessarily mean the change is unacceptable.

The proposed technical specification change does not involve any hardware changes nor does it affect the probability of any event initiators. There will be no change to normal plant operating parameters, engineered safety feature (ESF) actuation setpoints, accident mitigation capabilities, accident analysis assumptions or inputs.

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

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. As described in Section 9.1 of the WCAP–15049–A evaluation, the plant design will not be changed with this proposed technical specification CT increase. All safety systems still function in the same manner and there is no additional reliance on additional systems or procedures. The proposed accumulator CT increase has a very small impact on core damage frequency. The WCAP–15049–A evaluation demonstrates that the small increase in risk due to increasing the CT for an inoperable accumulator is within the acceptance criteria provided in RGs 1.174 and 1.177. No new accidents or transients can be introduced with the requested change and the likelihood of an accident or transient is not impacted.

The malfunction of safety related equipment, assumed to be operable in the accident analyses, would not be caused as a result of the proposed technical specification change. No new failure mode has been created and no new equipment performance burdens are imposed.

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

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

The proposed change does not involve a significant reduction in a margin of safety. There will be no change to the departure from nucleate boiling ratio (DNBR) correlation limit, the design DNBR limits, or the safety analysis DNBR limits.

The basis for the accumulator LCO, as discussed in Bases Section 3.5.1.1, is to ensure that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators, thereby providing the initial cooling mechanism during large RCS pipe ruptures. As described in Section 9.2 of WCAP–15049–A, the proposed change will allow plant operation with an inoperable accumulator for up to 24 hours, instead of 1 hour, before the plant would be required to begin shutting down. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to a large LOCA event, assuming accumulators are needed to mitigate the design basis event, has a very small impact on plant risk.

Since the frequency of a design basis large LOCA (a large LOCA with loss of offsite power) would be significantly lower than the large LOCA frequency of the WCAP–15049–A evaluation, the impact of increasing the accumulator CT from 1 hour to 24 hours on plant risk due to a design basis large LOCA would be significantly less than the plant risk increase presented in the WCAP–15049–A evaluation.

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

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

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

NRC Section Chief: Allen G. Howe.

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

Date of application request: December 8, 2003.

Description of amendment request: The licensee is proposing to revise Technical Specification (TS) Section 5.5.6, ''Containment Tendon Surveillance Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The proposed revision to TS 5.5.6 is to indicate that the Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and the applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC. The licensee has also proposed to delete the provisions of Surveillance Requirement 3.0.2 from this TS. In addition, the licensee is proposing to revise TS 5.5.16, ''Containment Leakage Rate Testing Program,'' to add exceptions to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Testing Program.''

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 revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations [as] required by Regulatory Guide 1.163, [because of the commitment in Appendix 3A of the Callaway Final Safety Analysis Report,] without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility.

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.

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces [of the containment]. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change does not involve a modification to the physical configuration of the plant (*i.e.,* no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent[s] that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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

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

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces [of the containment]. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The safety function of the containment as a fission product barrier will be maintained.

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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 30, 2003.

Description of amendment request: The amendment would revise the Technical Specifications (TSs) for alternating current (AC) sources—operating (TS 3.8.1) and electrical power distribution systems—operating (TS 3.8.9) by extending the required action completion times (CTs). For TS 3.8.1, the amendment would extend the CT to restore a single inoperable diesel generator (DG) to operable status by adding a note to the CT for Required Action B.4. A note would also be added to the CT for Required Action A.3 to restore a single inoperable offsite circuit to operable status to account for the note that would be added to the CT for Required Action B.4.

For TS 3.8.9, the CT for Required Action C.1 (to restore a single inoperable AC vital bus subsystem to operable status) would be extended to 24 hours. The second CTs, from the discovery of the failure to meet the limiting condition for operation (LCO), for Required Actions B.1 (to restore a single inoperable AC electrical power distribution subsystem to operable status), C.1 (given above), and D.1 (to restore a single inoperable direct current (DC) electrical power distribution subsystem to operable status) would be extended to 34 hours.

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

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

The proposed changes to the Completion Times do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the electrical power sources and distribution systems. The proposed changes to the second Completion Times are administrative in nature and only intended to prevent the plant from successively entering and exiting ACTIONS associated with different systems governed by one LCO without ever meeting the LCO. The electrical power sources and distribution subsystems will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by core damage frequency (CRF) and large early release frequency (LERF) is acceptable. In addition, for the Completion Time change, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are also acceptable. These changes are consistent with the acceptance criteria in Regulatory Guides 1.174 and 1.177. Therefore, since the electrical sources and distribution subsystems will continue to perform their [safety] functions with high reliability as originally assumed, and the increase in risk as measured by CDF, LERF, ICCDP, [and] ICLERP is acceptable, there will not be a significant increase in the consequences of any accidents [previously evaluated].

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended [safety] function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant [radiological] consequences.

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

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

The proposed changes do not result in a change in the manner in which the electrical distribution subsystems provide plant protection. The use of the Sharpe Station will provide an alternate AC power source in the event of emergent inoperability of a WCGS [Wolf Creek Generating Station] DG or a complete loss of all WCGS emergency AC power. The changes do not alter assumptions made in the safety analysis. The changes to Completion Times do not change any existing accident scenarios, nor create any new or different accident scenarios. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

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

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and is consistent with the acceptance criteria contained in Regulatory Guides 1.174 and 1.177.

Therefore, the proposed change[s do] not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: November 8, 2002.

Brief description of amendment: The amendment revised the Technical Specifications to delete the requirements for the auxiliary and fuel handling building air treatment system.

Date of issuance: December 12, 2003.

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

Amendment No.: 248.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 24, 2002 (67 FR 78517).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 28, 2003, as supplemented October 8, 2003.

Brief description of amendment: This amendment eliminates the need to credit Boraflex neutron-absorbing material for reactivity control in the spent fuel storage pool.

Date of issuance: December 22, 2003.

Effective date: December 22, 2003.

Amendment No.: 198.

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

Date of initial notice in ***Federal Register:*** July 8, 2003 (68 FR 40710). The October 8, 2003, supplement contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 22, 2003.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: December 11, 2002, as supplemented June 24, 2003.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) related to N–1 loop operation. Specifically, the changes eliminate N–1 loop operation from particular sections of the TSs and makes other changes that are clarifying and/or administrative in nature. In addition, the TS Bases are revised to address the proposed changes.

Date of issuance: December 10, 2003.

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

Amendment No.: 217.

Facility Operating License No. NPF–49: The amendment revised the TSs.

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2800). The January 24, 2003, supplement contained clarifying information and did not change the staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 10, 2003.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: March 24, 2003, as supplemented by letters dated June 25 and October 15, 2003.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to relocate certain reactor coolant system cycle-specific parameter limits from the TSs to the Core Operating Limits Report, and revises the minimum allowable reactor coolant system flow rate.

Date of issuance: December 19, 2003.

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

Amendment Nos.: 210 and 204.

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

Date of initial notice in Federal Register: September 18, 2003 (68 FR 54749), November 18, 2003 (68 FR 65090).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 19, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 19, 2003, as supplemented October 27, 2003.

Brief description of amendment: This amendment revises the Technical Specifications to require ''flow indication,'' rather than ''safety-grade flow indication,'' to satisfy Surveillance Requirement 4.7.1.7.e.2 for the motor driven feedwater pump.

Date of issuance: December 18, 2003.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 261.

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

Date of initial notice in ***Federal Register:*** June 10, 2003 (68 FR 34669).

The supplement dated October 27, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 2003.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: July 15, 2003.

Brief description of amendment: The amendment revises the Technical Specification (TS) requirements for surveillance of the status of Secondary Containment Isolation Valves and Blind Flanges in Surveillance Requirement 3.6.4.2.1, consistent with TS Task Force Traveler-45 Revision 2.

Date of issuance: December 5, 2003.

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

Amendment No.: 202.

Facility Operating License No. DPR–46: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 28, 2003 (68 FR 61479).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 5, 2003.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: April 11, 2003.

Brief description of amendment: The amendment revises TS Section 5.0, ''Administrative Controls,'' to make various administrative, editorial, and typographical changes.

Date of issuance: December 15, 2003.

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

Amendment No.: 213.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 12, 2003 (68 FR 64136).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 2003.

No significant hazards consideration comments received: No.

Southern California Edison Company, et al., Docket No. 50–206, San Onofre Nuclear Generating Station, Unit 1, San Diego County, California

Date of application for amendment: July 25, 2003, supplemented by letters dated October 3, 2003, and December 3, 2003.

Brief description of amendment: This amendment approves the use of the modified Unit 1 turbine gantry crane and turbine building support structure in a single failure proof application and at a rated capacity of 105 tons for handling of spent fuel casks as documented in the Defueled Safety Analysis Report (DSAR). The DSAR changes approved by this amendment are needed to permit use of the modified turbine gantry crane and turbine building support structure for lifting and handling of the spent fuel casks from the SONGS Unit 1 spent fuel pool to the Independent Spent Fuel Storage Installation.

Date of issuance: December 18, 2003.

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

Amendment No.: Unit 1–162.

Facility Operating License No.DPR–13: Amendment revises the license to permit use of the turbine building gantry crane in a single failure proof application at a rated capacity of 105 tons for handling of spent fuel casks.

Date of initial notice in ***Federal Register:*** September 18, 2003 (68 FR 54751).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 18, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 19, 2002, as supplemented October 20, 2003.

Brief description of amendments: These amendments correct various typographical, editorial, and other administrative errors currently in the Technical Specifications for Surry Power Station, Units 1 and 2.

Date of issuance: December 16, 2003.

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

Amendment Nos.: 238 and 237.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 4, 2003 (68 FR 5684). The supplement dated October 20, 2003, provided clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 16, 2003.

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, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By February 5, 2004, 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of the continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the petition for leave to intervene and request for hearing should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent 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).

STP Nuclear Operating Company, Docket No. 50–499, South Texas Project, Unit 2, Matagorda County, Texas

Date of amendment request: December 23, 2003.

Description of amendment request: The amendments revise Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' to extend the allowed outage time for Unit 2 Standby Diesel Generator 22 from 14 days to 21 days as a one-time change for the purpose of collecting data associated with failure of SDG–22.

Date of issuance: December 23, 2003.

Effective date: December 23, 2003.

Amendment Nos.: Unit No. 2: 148.

Facility Operating License Nos. NPF–76 and NPF–80: Amendments revise the Technical Specifications.

Public comments requested as to final no significant hazards consideration (NSHC): No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated December 23, 2003.

Attorney for licensee: A. H. Gutterman, Esquire, Morgan, Lewis & Bockius, LLP, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

Dated at Rockville, Maryland, this 24th day of December, 2003.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 04–8 Filed 1–5–04; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

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

Extension:

Rule 35d–1; SEC File No. 270–491; OMB Control No. 3235–0548.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*)(''Act'') the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for extension of the previously approved collection of information discussed below.

The title for the collection of information is ''Rule 35d–1 under the Investment Company Act of 1940, Investment Company Names.''

Rule 35d–1 under the Investment Company Act of 1940 [17 CFR 270.35d–1] generally requires that investment companies with certain names invest at least 80% of their assets according to what their names suggest. The rule provides that an affected investment company must either adopt this 80% requirement as a fundamental policy or adopt a policy to provide notice to shareholders at least 60 days prior to any change in its 80% investment policy. This preparation and delivery of the notice to existing shareholders is a collection of information within the meaning of the Act.

The Commission estimates that there are 7,200 open-end and closed-end management investment companies and series that have descriptive names that are governed by the rule. The Commission estimates that of these 7,200 investment companies, approximately 24 provide prior notice to their shareholders of a change in their investment policies per year. The Commission estimates that the annual burden associated with the notice requirement of the rule is 20 hours per affected investment company or series. The total burden hours for Rule 35d–1 is 480 per year in the aggregate (24 responses $\times$ 20 hours per response). Estimates of average burden hours are made solely for the purposes of the Act, and are not derived from a comprehensive or even a representative survey or study of the costs of Commission rules and forms.

The collection of information under Rule 35d–1 is mandatory. The information provided under Rule 35d–1 is not kept confidential. The Commission may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

Written comments are invited on: (a) Whether the proposed collection of information is necessary for the proper performance of the functions of the agency, including whether the information will have practical utility; (b) the accuracy of the agency's estimate of the burden of the collection of information; (c) ways to enhance the quality, utility, and clarity of the information collected; and (d) ways to minimize the burden of the collection of information on respondents, including through the use of automated collection techniques or other forms of information technology. Consideration will be given to comments and suggestions submitted in writing within 60 days of this publication.

Please direct your written comments to Kenneth A. Fogash, Acting Associate Executive Director/CIO, Office of Information Technology, Securities and Exchange Commission, 450 5th Street, NW., Washington, DC 20549.

Dated: December 23, 2003.

Jill M. Peterson,

Assistant Secretary.

[FR Doc. 04–152 Filed 1–5–04; 8:45 am]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension:

Rule 15c2–12; SEC File No. 270–330; OMB Control No. 3235–0372.

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 collection of information summarized below. The Commission plans to submit this existing collection of information to the Office of Management and Budget for extension and approval.

- Rule 15c2–12 Disclosure requirements for municipal securities.

Rule 15c2–12, under the Securities Exchange Act of 1934, requires underwriters of municipal securities: (1) To obtain and review a copy of an official statement deemed final by an issuer of the securities, except for the omission of specified information; (2) in non-competitively bid offerings, to make available, upon request, the most recent preliminary official statement, if any; (3) to contract with the issuer of the securities, or its agent, to receive, within specified time periods, sufficient copies of the issuer's final official statement to comply both with this rule and any rules of the MSRB; (4) to provide, for a specified period of time, copies of the final official statement to any potential customer upon request; (5) before purchasing or selling municipal securities in connection with an offering, to reasonably determine that the issuer or other specified person has undertaken, in a written agreement or contract, for the benefit of holders of such municipal securities, to provide certain information about the issue or issuer on a continuing basis to a nationally recognized municipal securities information repository; and (6) to review the information the issuer of the municipal security has undertaken to provide prior to recommending a transaction in the municipal security.

These disclosure and recordkeeping requirements will ensure that investors have adequate access to official disclosure documents that contain details about the value and risks of particular municipal securities at the time of issuance while the existence of