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.* The application dated December 19, 2003, can be accessed under ADAMS Accession No. ML033570112.

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, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland this 13th day of January 2004.

For the Nuclear Regulatory Commission.

John G. Lamb,

Project Manager, Section 1, Project Directorate III, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 24, 2003, through January 8, 2004. The last biweekly notice was published on January 6, 2003 (69 FR 691).

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 19, 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, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: May 1, 2003.

Description of amendment request: The proposed amendment would revise the Clinton Power Station (CPS) Technical Specifications to (1) support an expansion of the core flow operating range, (2) implement an Oscillation Power Range Monitor (OPRM) Instrumentation system, and (3) implement the Detect and Suppress Solution—Confirmation Density approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities. These changes will support operation at 3,473 megawatts thermal with core flow as low as 85 percent of rated core flow. The expanded operating range is identified as Maximum Extended Load Line Limit Analysis Plus (MELLLA+). The scope of evaluations required to support the expansion of the core flow operating range to MELLLA+ boundary is contained in the General Electric Licensing Topical Report (LTR) NEDC–33006P, ''Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report.''

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

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

Response: No.

The probability (frequency of occurrence) of a design basis accident (DBA) occurring is not affected by the operating range expansion, because the plant continues to comply with the regulatory and design basis criteria established for plant equipment. The MELLLA+ core operating range expansion does not require significant plant hardware modifications. The core operating range expansion involves changes to the operating power-to-flow map and a small number of setpoints and alarms. Because there is no change in the operating pressure, power, steam flow rate, or feedwater flow rate, there are no significant effects on the plant hardware outside of the Nuclear Steam Supply System (NSSS). The MELLLA+ operating range expansion does not cause additional requirements to be imposed on any of the safety, balance-of-plant, electrical, or auxiliary systems. No changes to the power generation and electrical distribution systems are required due to the introduction of MELLLA+. An evaluation of the probabilistic safety assessment concludes that the calculated increase in core damage frequencies due to the MELLLA+ operating range expansion are very small. Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to the MELLLA+ operating range expansion. No new challenges to safety-related equipment result from the MELLLA+ operating range expansion. As a result, there is no significant increase in the probability of an accident previously evaluated.

The proposed changes specify limiting conditions for operation, required actions and surveillance requirements for the OPRM system, and allows operation in regions of the power-to-flow map currently restricted by the requirements of the Interim Corrective Actions (ICAs) and certain limiting conditions of operation of TS Section 3.4.1. The restrictions of the ICAs and TS Section 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of thermal-hydraulic oscillations that may develop into a thermal-hydraulic instability event. A thermal-hydraulic instability event has the potential to challenge the Minimum Critical Power Ratio (MCPR) safety limit. The OPRM system can automatically detect and suppress conditions necessary for thermal-hydraulic instability. The Backup Stability Protection (BSP), in lieu of the ICAs, will provide adequate protection should the OPRM equipment become temporarily inoperable. With the activation of the OPRM system, the restrictions of the ICAs and TS Section 3.4.1 will no longer be required.

The probability of a thermal-hydraulic instability event is impacted by power to flow conditions such that only during operation inside specific regions of the power-to-flow map, in combination with power shape and inlet enthalpy conditions, can the occurrence of an instability event be postulated to occur. Operation in these regions may increase the probability that operation with conditions necessary for a thermal-hydraulic instability can occur.

When the OPRM is operable, the OPRM can automatically detect the imminent onset of power oscillations and generate a trip signal. Actuation of a Reactor Protection System (RPS) trip will suppress conditions necessary for thermal-hydraulic instability and decrease the probability of a thermal-hydraulic instability event. In the event the trip capability of the OPRM is not maintained, the proposed changes limit the period of time before an alternate method to detect and suppress thermal-hydraulic oscillations is required. Since the duration of this period of time is limited, the increase in the probability of a thermal-hydraulic instability event is not significant. Therefore, the proposed changes do not result in a significant increase in the probability of an accident previously evaluated.

The DSS–CD solution is designed to identify power oscillations upon inception and initiate control rod insertion (*i.e.*, scram) to terminate the oscillations prior to any significant amplitude growth. The DSS–CD provides protection against violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for anticipated oscillations. Compliance with Criterion 10, ''Reactor design.'', and Criterion 12, ''Suppression of reactor power oscillations.'', of 10CFR50, Appendix A, ''General Design Criteria For Nuclear Power Plants,'' is accomplished via an automatic action. A developing instability event is suppressed by the DSS–CD system with substantial margin to the SLMCPR and no clad damage, with the event terminating in a scram and never developing into an accident. The DSS–CD system does not interact with equipment whose failure could cause an accident. Scram setpoints in the DSS–CD will be established so that analytical limits are met. The reliability of the DSS–CD will meet or exceed that of the existing system. No new challenges to safety-related equipment will result from the DSS–CD solution. Because an instability event would reliably terminate in an early scram without impact on other safety systems, there is no significant increase in the probability of an accident.

The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and SLMCPR continue to be met. The fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO–24011, ''GESTAR II,'' (Reference 12). Challenges to fuel are evaluated, and shown to still meet the criteria of 10 CFR 50.46, ''Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors.'', 10 CFR 50 Appendix K, ''ECCS Evaluation Models,'' and Regulatory Guide 1.70, ''Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,'' Section 6.3. Challenges to the containment have been evaluated, and the containment and its associated cooling systems meet Criterion 38, ''Containment heat removal.'', and Criterion 50, ''Containment design basis.'', of the general design criteria. Radiological release events have been evaluated, and are shown to be below the regulatory limits of 10 CFR 100, ''Reactor Site Criteria''. Operation in the MELLLA+ region does not result in an increase in the consequences of an accident previously evaluated. Operation within the MELLLA+ region has been evaluated to ensure that the CPS response to accidents and transients remains within acceptable criteria. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

An unmitigated thermal-hydraulic instability event is postulated to cause a violation of the MCPR safety limit. The proposed changes ensure mitigation of thermal-hydraulic instability events prior to challenging the MCPR safety limit if initiated from anticipated conditions by detection of the onset of oscillations and actuation of an RPS trip signal when the OPRM system is operable. The OPRM also provides the capability of an RPS trip being generated for thermal-hydraulic instability events initiated from unanticipated but postulated conditions. These mitigative capabilities of the OPRM system would become available as a result of the proposed changes and have the potential to reduce the consequences of unanticipated and postulated thermal-hydraulic instability events.

As stated above, the DSS–CD solution meets the requirements of Criterion 10 and Criterion 12 of the GDC by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel SLMCPR. Proper operation of the DSS–CD system does not affect any fission product barrier or Engineered Safety Feature. Thus, the proposed change cannot change the consequences of any accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability 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 previously evaluated?

Response: No.

Equipment that could be affected by MELLLA+ has been evaluated and no new operating mode, safety related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations, defined in the CPS Updated Safety Analysis Report (USAR), has been evaluated, and no new or different kind of accident has been identified. The MELLLA+ operating range expansion uses existing technology and NRC approved safety analysis methodology, and applies them within the capabilities of already existing plant equipment in accordance with presently existing regulatory and industry criteria. The MELLLA+ operating range expansion will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes specify limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power-to-flow map currently restricted by the requirements of the ICAs and TS Section 3.4.1. The OPRM system uses input signals shared with the Average Range Power Monitor (APRM) system and rod block functions to monitor core conditions and generate an RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of the other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. No new failure modes of either the new OPRM equipment or of the existing APRM equipment have been introduced. Therefore, operation with the OPRM system does not create the possibility of a new or different kind of accident from any previously evaluated.

The DSS–CD solution operates within the existing Option III OPRM hardware. Implementation of the DSS–CD will require a software/hardware change to the existing Option III system. No new operating mode, safety-related equipment lineup, accident scenario, system interaction, or equipment failure mode was identified. Therefore, the DSS–CD solution will not adversely affect plant equipment. Because there are no significant hardware changes, there is no change in the possibility or consequences of a failure. The worst-case failure of the equipment is a failure to initiate mitigating action (*i.e.*, scram), but no failure can cause an accident of a new or different kind than any previously evaluated.

As such the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The calculated loads on all affected structures, systems and components have been shown to remain within design allowables for all design basis event categories. No NRC acceptance criteria are exceeded. The margins of safety currently included in the design of the plant are not affected by the MELLLA+ operating range expansion. Because the plant configuration and response to transients and hypothetical accidents do not result in exceeding the presently approved NRC acceptance limits, operation in the MELLLA+ region does not involve a significant reduction in a margin of safety.

The OPRM system monitors small groups of LPRM signals for indication of local variations of core power consistent with thermal-hydraulic oscillations and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated thermal-hydraulic instability event has the potential to result in a challenge to the MCPR safety limit. The OPRM system provides the capability to automatically detect and suppress conditions which might result in a thermal-hydraulic instability event and thereby maintains the margin of safety by providing automatic protection for the MCPR safety limit while reducing the burden on the control room operators significantly. The BSP, in lieu of the ICAs, will provide adequate protection should the OPRM equipment become temporarily inoperable. Operation with the OPRM system does not involve a significant reduction in a margin of safety.

The DSS–CD solution is designed to identify the power oscillations upon inception and initiate control rod insertion to terminate (*i.e.*, scram) the oscillations prior to any significant amplitude growth. The DSS–CD solution algorithm will maintain or increase the margin to the SLMCPR for anticipated instability events. The safety analyses in NEDC–33075P demonstrate the margin to the SLMCPR for postulated bounding stability events. In addition, the current Option III algorithms are retained to provide defense-in-depth protection for unanticipated reactor instability events. As a result, there is no impact on the MCPR Safety Limit identified for an instability event.

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

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

Attorney for licensee: Edward J. Cullen, Deputy General Counsel Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 23, 2003.

Description of amendment request: The licensee proposed to revise Section 3.4.A and 3.5.A.2 of the Technical Specifications to clarify requirements for inoperable components and allow meeting the water availability requirements during periods of core spray system inoperability (*e.g.*, when the plant is shutdown) in an alternate manner. Specifically, this would allow the required water volume for core spray system operability be located in the torus, condensate storage tank, or a combination of both, in order to provide operational flexibility in water management and outage work scheduling. Additionally, the licensee proposed to improve consistency of verification requirements within the specifications and provide more definitive bases for the specifications. No physical changes to the plant are involved, and the requirements in the current specifications will be maintained.

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

The first standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes will be made in a manner such that the current requirements are maintained for the core spray system. The source of core spray water was not considered as a precursor of any previously analyzed and evaluated accident. No hardware design change is involved with the proposed amendment. Thus, the proposed amendment would create no adverse effect on the functional performance of any plant structure, system, or component (SSC). All SSCs will continue to perform their design functions with no decrease in their capabilities to mitigate the previously analyzed consequences of postulated accidents. Accordingly, the revised specifications will lead to no increase in the consequences of an accident previously evaluated, and no increase of the probability of an accident previously evaluated.

The second standard requires that operation of the unit in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment is not the result of a hardware design change, nor does it lead to the need for a hardware design change. There is no change in the methods the unit is operated. As a result, all SSCs will continue to perform as previously analyzed by the licensee, and previously evaluated and accepted by the NRC staff. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

The third standard requires that operation of the unit in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. Since the licensee did not propose to exceed or alter a design basis or safety limit, and did not propose to operate any component in a less conservative manner, the proposed amendment will not affect in any way the performance characteristics and intended functions of any SSC. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: December 23, 2003.

Brief description of amendments: The licensee proposed to revise various parts of the Technical Specifications (TSs) to allow entry into a mode or other specified condition in the applicability of a specification while in a condition statement and the associated required actions of the TSs, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Specifically, TS 3.0, ''Limiting Conditions for Operation (General),'' as well as other portions of the TSs (*i.e.*, Sections 3.4, 3.7, and 3.8) referencing TS 3.0, will be revised.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). In its application for amendment, the licensee affirmed the applicability of the following NSHC determination.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee presented an analysis of NSHC by endorsing the model NSHC published in 68 FR 16579 (reproduced 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 allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change 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). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Richard J. Laufer.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–317, Calvert Cliffs Nuclear Plower Plant, Unit No. 2, Calvert County, Maryland

Date of amendment request: September 30, 2003.

Description of amendment request: The proposed amendment would increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 2 spent fuel pool by taking credit for soluble boron, burnup and configuration control in maintaining acceptable margins of subcriticality.

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

The proposed change will increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 2 spent fuel pool (SFP) by taking credit for soluble boron, burnup and configuration control in maintaining acceptable margins of subcriticality. The proposed change will modify Technical Specification 4.3.1 ''Criticality,'' add Technical Specification 3.7.16, ''Spent Fuel Pool Boron Concentration'' and add Technical Specification 3.7.17 ''Spent Fuel Pool Storage.'' The postulated accidents for the SFP are basically four types; (1) dropped fuel assembly on top of the storage rack, (2) a misloading accident, (3) an abnormal location of a fuel assembly, and (4) loss-of-normal cooling to the SFP.

There is no increase in the probability of a fuel assembly drop accident in the SFP when considering the higher enriched fuel or the presence of soluble boron in the SFP water. Dropping a fuel assembly on top of the SFP storage racks is not credible at Calvert Cliffs due to the design of the spent fuel handling machine and the height of the SFP storage racks. The handling of fuel assemblies has always been performed in borated water and will not change as a result of crediting soluble boron in the SFP criticality analysis. The proposed change does not change the general design or characteristics of the fuel assemblies. Therefore, the proposed change does not increase the probability of a fuel assembly drop accident.

There is no increase in the probability of the accidental misloading of irradiated fuel assemblies into the SFP storage racks when considering the higher enriched fuel or the presence of soluble boron in the SFP water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures.

Due to the design of the SFP storage racks, an abnormal placement of a fuel assembly into the SFP storage racks is not possible. Also, the design of the SFP prevents an inadvertent placement of a fuel assembly between the outer most storage cell and the pool wall. The proposed change does not make any change to the design of SFP. Therefore, there is no increase in the probability of abnormal placement of a fuel assembly into the SFP storage racks.

The proposed change will not result in any changes to the SFP cooling system, and the fuel assembly design and characteristics are not changed by an increase in fuel enrichment. Therefore, there is no increase in the probability of a loss of SFP cooling. Also, since a high concentration of soluble boron has always been maintained in the SFP water, there is no increase in the probability of the loss of normal cooling to the SFP water considering the presence of soluble boron in the pool water for criticality control.

There is no increase in the consequences of an accidental drop, accidental misloading, or abnormal placement of a maximum enriched fuel assembly into the SFP storage racks, because the criticality analysis demonstrates that the pool will remain subcritical following either event. The Technical Specification limit for SFP boron concentration will ensure that an adequate SFP boron concentration will be maintained.

There is no increase in the consequences of a loss-of-normal SFP cooling because the Technical Specification boron concentration provides significant negative reactivity. Loss of the SFP water via boiling will not result in a loss of soluble boron, since the soluble boron is not volatile. Therefore, loss of SFP cooling system, without makeup flow, is not a mechanism for boron dilution. Even in the unlikely event that soluble boron in the SFP is completely diluted via unborated makeup flow, a pool completely filled with maximum enriched unburned assemblies will remain subcritical by a design margin that meets the requirements of 10 CFR 50.68.

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 will increase the maximum enrichment limit of the fuel assemblies that can be stored in the Unit 2 SFP by taking credit for soluble boron, burnup and configuration control in maintaining acceptable margins of subcriticality. Increasing the maximum enrichment limit does not create a new type of criticality accident.

Soluble boron has been maintained in the SFP water and is currently required by procedures. Therefore, crediting soluble boron in the SFP criticality analysis will have no effect on normal pool operation and maintenance. Crediting soluble boron will only result in increased sampling to verify the boron concentration in accordance with the proposed Technical Specification Surveillance Requirement. This increased sampling will not create the possibility of a new or different kind of accident.

A dilution of the SFP soluble boron has always been a possibility. However, the boron dilution event previously had no consequences, since boron was not previously credited in the accident analysis. The initiating events that were considered for having the potential to cause dilution of the boron in the SFP to a level below that credited in the criticality analyses fall into three categories: dilution by flooding, dilution by loss-of-coolant induced makeup, and dilution by loss-of-cooling system induced makeup. The SFP dilution analysis demonstrates that a dilution event that could increase k-effective in the SFP to greater than 0.95 is not a credible event. It is not credible that dilution could occur for the required length of time without operator notice, since this event would activate the high level alarm and initiate Auxiliary Building flooding. In addition, in excess of 1,043,000 gallons of unborated water must be added to the SFP to reach the minimum soluble boron concentration. This is more water volume than is contained in both pretreated water storage tanks and also more water volume than is contained in the demineralized water storage tank and both condensate storage tanks combined. Even in the unlikely event that soluble boron in the SFP is completely diluted, the SFP will remain subcritical by a design margin that meets the requirements of 10 CFR 50.68.

Burned assemblies have been stored in the SFP for many cycles. Therefore, crediting burnup in the SFP criticality analysis will have no effect on normal pool operation and maintenance. Fuel assembly placement, although more complex, will continue to be controlled pursuant to approved fuel handling procedures and in accordance with Technical Specification spent fuel rack storage configuration limitations.

The proposed change will not result in any other change in the plant configuration or equipment design. 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 Technical Specification changes proposed by this license amendment request will provide an adequate safety margin to ensure that the stored fuel assembly array of maximum enriched fuel will always remain subcritical. Those limits are based on a plant specific criticality analysis performed for the Calvert Cliffs Unit 2 SFP, that include technically supported margins.

Soluble boron is used to provide subcritical margin such that the SFP k-effective is maintained less than or equal to 0.95. Since k-effective is less than or equal to 0.95, the current margin of safety is maintained. In addition, while the criticality analysis utilized credit for soluble boron, the fuel in the SFP rack will remain subcritical with no soluble boron with a 95 percent probability at a 95 percent confidence level as required by 10 CFR 50.68. This substantial reduction in the SFP soluble boron concentration was evaluated and shown not to be credible.

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 amendments request involves no significant hazards consideration.

Attorney for licensee: James M. Petro, Jr., Esquire, Counsel, Constellation Energy Group, Inc., 750 East Pratt Street, 5th floor, Baltimore, MD 21202.

NRC Section Chief: Richard J. Laufer.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: October 16, 2003.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) 3.4.9 to change the minium pressurizer (PZR) heater capacity from 126 to 400 kW to correct a non-conservative TS associated with a PZR design basis deficiency.

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

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

No. The proposed changes revise the minimum PZR [pressurizer] heater capacity required and capable of being powered from an emergency power supply source. UFSAR [Updated Final Safety Analysis Report] do not take credit for PZR heater operation; however, an implicit initial condition assumption of the safety analyses is that RCS [Reactor Coolant System] is operating at normal pressure. Assurance of this assumption is enhanced due to these proposed changes. Consequently, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. These changes correct a non-conservative value from the TS [technical specification] and are necessary to assure RCS pressure control and adequate natural circulation cooling. The available heater capacity being powered from an emergency power supply is approximately 1000 kW for the most restrictive unit which exceeds the proposed 400 kW minimum capacity required by TS. The proposed changes help ensure that the RCS is operating at normal pressure which is an implicit initial assumption used in several UFSAR described safety analyses. Consequently, these changes do not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

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

No. The proposed change does not adversely affect any plant safety limits, set points, or design parameters. The change also does not adversely affect the fuel, fuel cladding, RCS, or containment integrity. Therefore, the proposed changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: December 5, 2003.

Description of amendment request: The proposed amendment would revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) values in Technical Specification 1.1.A.1 to incorporate the results of the cycle-specific core reload analysis for Vermont Yankee Nuclear Power Station Cycle 24 operation.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The basis of the Safety Limit Minimum Critical Power Ratio (SLMCPR) is to ensure no mechanis0tic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and probability of fuel damage is not increased. The derivation of the revised SLMCPR for Vermont Yankee for incorporation into the Technical Specifications, and its use to determine plant and cycle-specific thermal limits, have been performed using NRC [U.S. Nuclear Regulatory Commission] approved methods. These plant-specific calculations are performed each operating cycle and if necessary, will require future changes to these values based upon revised core designs. The revised SLMCPR values do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

Based on the above, Vermont Yankee has concluded that the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes result only from a specific analysis for the Vermont Yankee core reload design. These changes do not involve any new or different methods for operating the facility. No new initiating events or transients result from these changes.

Based on the above, Vermont Yankee has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The new SLMCPR is calculated using NRC approved methods with plant and cycle specific parameters for the current core design. The SLMCPR value remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle specific transients are evaluated. Accordingly, the margin of safety is maintained with the revised values.

As a result, Vermont Yankee has determined that the proposed change will not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: Darrell J. Roberts, Acting.

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

Date of amendment request: December 19, 2003.

Description of Amendment Request: The proposed amendment deletes requirements from the Technical Specifications (TS) to maintain hydrogen recombiners and hydrogen monitors.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50374), on possible amendments to eliminate the hydrogen recombiners from TS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the Consolidated Line Item Improvement Process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated December 19, 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 revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG [Regulatory Guide] 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3—hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

Based upon the reasoning presented above, the requested change does not involve a significant hazards consideration. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, Docket No. STN 50–454, Byron Station, Unit 1, Ogle County, Illinois

Date of amendment request: December 5, 2003.

Description of amendment request: The proposed amendment would allow irradiation of two lead test assemblies (LTAs) and two ''standard'' Westinghouse 17x17 VANTAGE+ZIRLO™ assemblies beyond the current fuel rod-average licensing basis burnup value of 60,000 MWD/MTU up to 65,000 MWD/MTU during the current operating cycle (B1C13). Irradiation of these four assemblies is intended to confirm the acceptable use of the ZIRLO™ alloys to a discharge burnup level exceeding the current licensing basis.

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

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

Fuel rod defects or failures are not considered as initiators for any previously analyzed accident; therefore the requested license amendment will have no effect on the probability of any previously evaluated accident. In addition, NRC-approved methodologies and technical reports have been used in the B1C13 specific reload safety evaluation to confirm that the fuel rod design limits will be met; therefore, increasing the burnup limit of the specified fuel assemblies to the requested value will not increase the consequences of any previously analyzed accident.

The regular ZIRLO™ and ZIRLO™ (LT–1) high burnup fuel rods will continue to satisfy the specified acceptable fuel design limits (SAFDLs) specified in NRC-approved Westinghouse topical reports. The clad integrity of the ZIRLO™ and ZIRLO™ (LT–1) high burnup rods will be maintained as the subject fuel assemblies will be placed in less than limiting core locations and will continue to meet the safety parameter requirements. The acceptability of using the ZIRLO™ and ZIRLO™ (LT–1) high burnup rods has been evaluated and confirmed in the B1C13 Reload Safety Evaluation supported by the Westinghouse LTA Report, ''Byron Unit 1 Cycle 13 LTA Report,'' dated August 2003.

It has been shown in WCAP–12610–P–A, that even though there are variations in core inventories of isotopes due to extended burnup up to 75,000 MWD/MTU, there are no significant increases of isotopes that are major contributors to accident doses. It is worthy to note that, at higher burnups, there is actually a reduction in certain isotopes that are major dose contributors under accident situations (*e.g.*, Kr-88). With only a limited number of ZIRLO™ and ZIRLO™ (LT–1) high burnup rods in the entire core, any variation of isotopes will be extremely small. Thus, the radiation dose limitations of 10 CFR 100, ''Reactor Site Criteria,'' will not be exceeded.

Based on the above discussion, it is concluded that 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 to increase the current fuel rod-average burnup limit does not involve the use or installation of new equipment and all currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed change will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

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

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

The proposed change to increase the current fuel rod-average burnup limit of 60,000 MWD/MTU up to 65,000 MWD/MTU during B1C13 will cause the following fuel rod design criteria to become more limiting: Fuel rod growth, clad fatigue, rod internal pressure and cladding corrosion. However, the regular ZIRLO™ and ZIRLO™ (LT–1) high burnup fuel rods will continue to satisfy the SAFDLs specified in NRC-approved Westinghouse topical reports as noted above. The clad integrity of the ZIRLO™ and ZIRLO™ (LT–1) high burnup rods and the appropriate margin to safety will be maintained as the subject fuel assemblies will be placed in less than limiting core locations and will continue to meet the safety parameter requirements. The acceptability of using the ZIRLO™ and ZIRLO™ (LT–1) high burnup rods has been evaluated and confirmed in the B1C13 Reload Safety Evaluation supported by the Westinghouse LTA Report, ''Byron Unit 1 Cycle 13 LTA Report,'' dated August 2003.

Based on the above evaluation, 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, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, Docket No. 50–265, Quad Cities Nuclear Power Station, Unit 2, Rock Island County, Illinois Date of amendment request:

Date of amendment request: November 14, 2003, as supplemented by letter dated December 23, 2003.

Description of amendment request: The proposed amendment would revise the values and wording of the technical specifications safety limit minimum critical power ratio (SLMCPR).

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for Quad Cities Nuclear Power Station (QCNPS), Unit 2, Cycle 18 such that the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences (AOOs).

Changing the SLMCPR does not increase the probability of an evaluated accident. The 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.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during any transients or anticipated operational occurrences. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criterion (*i.e.*, that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, 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. Does the proposed change 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 creating one or more new precursors of that accident. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for QCNPS, Unit 2, Cycle 18.

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

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

The SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criteria (*i.e.*, that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) are met.

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

The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The 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, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: December 2, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications to allow a reduction in the minimum reactor coolant system flow, corresponding to an increase in the steam generator tube plugging limit from 15 percent to 30 percent.

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

(1) Operation of the facility in accordance with the proposed amendment would involve a significant reduction in the margin of safety .

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

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

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: October 8, 2003.

Description of amendment request: The proposed amendment is to revise Technical Specifications (TS) 4.2.b.3.a, ''Inspection Frequency,'' for the Kewaunee Nuclear Power Plant (KNPP). The proposed one-time change would revise the steam generator (SG) inspection interval requirements in TS for KNPP to allow a 40-month inspection interval after one SG inspection.

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

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

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in Technical Specifications (TS) 4.2.b.3.a, following the Kewaunee Nuclear Plant, spring 2003 refueling outage, to allow a 40-month inspection frequency after one inspection, rather than after two consecutive inspections results that are within the C–1 category.

The proposed on-time extension of the SG tube in-service inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Kewaunee Updated Safety Analysis Report (USAR). As such, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would be very low. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last KNPP refueling outage (*i.e.*, the first refueling outage following Steam generator replacement (SGR) exceeded the TS requirements for the first two refuleing outages after SGR. That is, more tubes were inspected than were required by TS (*i.e.*, 100 percent inspection was performed). Currently, KNPP does not have an active SG damage mechanism, and will meet the current industry examination guidelines without performing additional SG inspections until the spring 2006 refueling outage. Additionally, as part of our SG Tube Surveillance Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97–06, ''Steam Generator Program Guidelines,'' performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the KNPP spring 2003 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

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

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

The proposed change revises the SG inspection frequency requirements in TS 4.2.b.3.a, to allow a 40-month inspection interval after one inspection, rather than after two consecutive inspections with inspection results within the C–1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections (*i.e.*, 100 percent) performed during the last KNPP refueling outage (*i.e.*, the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary to secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube in-service inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

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

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

The SG tubes are an integral part of the Reactor coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SG is maintained by ensuring integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SG. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for similar replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

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

NRC Section Chief: L. Raghavan.

Nuclear Management Company, LLC, Docket No. 50–282, Prairie Island Nuclear Generating Plant, Unit 1, Goodhue County, Minnesota

Date of amendment request: August 27, 2003, as supplemented December 16, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification 5.5.14, ''Containment Leakage Rate Testing Program,'' to allow Unit 1 to be excepted from the requirements of Regulatory Guide 1.163, for post-modification integrated leakage rate testing associated with steam generator replacement.

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

Response: No.

The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

Integrated leak rate tests are performed to assure the leak-tightness of the primary containment boundary system, and as such they are not accident initiators. Therefore, not performing an integrated leak rate test will not affect the probability of an accident previously evaluated.

The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. For the Unit 1 steam generator replacement modification, this intent will be satisfied by performing the American Society of Mechanical Engineers code required inspections and tests. Since the leak-tightness integrity of the primary containment boundary affected by replacement of the steam generators will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident.

Therefore adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

Providing an exception from performing a test does not involve a physical change to the plant nor does it change the operation of the plant. Thus it cannot introduce a new failure mode.

Therefore adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements 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 proposed change would provide the Prairie Island Nuclear Generating Plant an exception from performing a required containment integrated leak rate test following the replacement of the steam generators in Unit 1.

The intent of post-modification integrated leak rate testing requirements is to assure the leak-tight integrity of the area affected by the modification. This intent will be satisfied by performing American Society of Mechanical Engineers code required inspections and tests. The acceptance criterion for American Society of Mechanical Engineers code system pressure testing for the base metal and welds is no leakage. In addition, the test pressure for the system pressure test will be several times that required during an integrated leak rate test. Since the leak-tight integrity of the primary containment boundary affected by replacement of the steam generators will be assured, there is no change in the primary containment boundary's ability to confine radioactive materials during an accident.

Therefore, adding a Technical Specification requirement that provides an exception for Unit 1 from the steam generator replacement post-modification integrated leak rate testing requirements 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 requests involve 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: L. Raghavan.

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 22, 2003.

Description of amendment request: The proposed amendment would revise the Unit 1 and 2 Technical Specifications (TSs) by adding TS 3.3.1.3, ''Oscillation Power Range Monitor (OPRM) Instrumentation,'' and revising TS 3.4.1, ''Recirculation Loops Operating,'' and TS 5.6.5, ''Core Operating Limits Report,'' to remove specifications and information related to current stability specifications which will no longer be needed.

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 OPRM most directly affects the APRM [average power range monitor] and LPRM [local power range monitor] portions of the Power Range Neutron Monitoring system. Its installation does not affect the operation of these sub-systems. None of the accidents or equipment malfunctions affected by these sub-systems are affected by the presence or operation of the OPRM. The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux changes. The APRM Fixed Neutron Flux-High function is capable of generating a trip signal to prevent fuel damage or excessive reactor pressure. For the ASME [American Society of Mechanical Engineers] overpressurization protection analysis in FSAR [Final Safety Analysis Report] Chapter 5, the APRM Fixed Neutron Flux-High function is assumed to terminate the main steam isolation valve closure event. The high flux trip, along with the safety/relief valves, limits the peak reactor pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis in Chapter 15 takes credit for the APRM Fixed Neutron Flux-High function to terminate the CRDA. The Recirculation Flow Controller Failure event (pump runup) is also terminated by the high neutron flux trip. The APRM Fixed Neutron Flux-High function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the Safety Limits (*e.g.,* MCPR [minimum critical power ratio] and Reactor pressure) being exceeded.

The installation of the OPRM equipment does not increase the consequences of a malfunction of equipment important to safety. The APRM and RPS [Reactor Protection System] systems are designed to fail in a tripped (fail safe) condition; the OPRM will have no affect on the consequences of the failure of either system. An inoperative trip signal is received by the RPS any time an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs. These functions are not specifically credited in the accident analysis, but are retained for the RPS as required by the NRC approved licensing basis.

The OPRM allows operation under operating conditions presently restricted by the current Technical Specifications by providing automatic suppression functions in the area of concern in the event an instability occurs. The consequences of any accident or equipment malfunction are not increased by operating under those conditions. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural core circulation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system.

This change does not increase the probability of an accident as previously evaluated. The OPRM is designed and installed to not degrade the existing APRM, LRPM, and RPS systems. These systems will still perform all of their intended functions. The new equipment is tested and installed to the same or more restrictive environmental and seismic envelopes as the existing systems. The new equipment has been designed and tested to electromagnetic interference (EMI) requirements which assure correct operation of the existing equipment. The new system has been designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The electrical loading is within the capability of the existing power sources and the heat loads are within the capability of existing cooling systems. The OPRM allows operation under operating conditions presently forbidden or restricted by the current Technical Specifications. No other transient or accident analysis assumes these operating restrictions.

Therefore, this 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.

This proposal does not create the possibility of a new or different type of accident from any accident previously evaluated. The OPRM system is a monitoring and accident mitigation system that cannot create the possibility for an accident not previously evaluated.

The OPRM will allow operation in conditions restricted by the current Technical Specifications. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural circulation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system. No new failure modes of either the new OPRM equipment or of the existing APRM equipment have been introduced. Quality software design, testing, implementation and module self-health testing provides assurance that no new equipment malfunctions due to software errors are created. The possibility of an accident of a new or different type than any evaluated previouly is not created.

The new OPRM equipment is designed and installed to the same system requirements as the existing APRM equipment and is designed and tested to have no impact on the existing functions of the APRM system. Appropriate isolation is provided where new interconnections between redundant separation groups are formed. The OPRM modules have been designed and tested to assure that no new failure modes have been introduced.

Therefore, the proposed change 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.

There has been no reduction in the margin of safety as defined in the basis for the Technical Specifications. The OPRM system does not negatively impact the existing APRM system. As a result, the margins in the Technical Specifications for the APRM system are not impacted by this addition.

Current operation under the ICAs [interim corrective actions] provides an acceptable margin of safety in the event of an instability event as the result of preventive actions and Technical Specification controlled response by the control room operators. The OPRM system provides an increase in the reliability of the protection of the margin of safety by providing automatic protection of the MCPR safety limit, while the protection burden is significantly reduced for the control room operators. This protection is demonstrated as described above, and in the NRC reviewed and approved Topical Reports NEDO–32465–A and CENPD–400–P–A.

Replacement of the ICA operating restrictions from Technical Specifications with the OPRM system does not affect the margin of safety associated with any other system or fuel design parameter.

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

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

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.

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

Date of amendment request: November 17, 2003.

Description of amendment request: The proposed change would revise the Technical Specifications to delete the primary containment isolation valves and instrumentation associated with the permanent removal of the reactor vessel head spray piping.

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 not involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed changes to Technical Specification Tables 3.3.2–1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 do not involve a change in structures, systems, or components that would affect the probability or consequences of any accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report.

The proposed changes involve eliminating piping and valves associated with the reactor head spray. The reactor head spray system was initially provided to cool down the steam dryer and separator during shutdown. The head spray system is not credited for the prevention or mitigation of any accident. Therefore, neither the offsite or control room radiological consequences are affected. The head spray piping removal and addition of a bolted flange on the reactor coolant pressure boundary enhances plant safety by eliminating a source of pipe whip and potential leakage. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be periodically tested in conjunction with the containment integrated leak rate test.

Therefore, as discussed above, this modification does not involve a significant increase in the probability or consequences from any accident previously analyzed.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No.

The proposed changes to Technical Specification Tables 3.3.2–1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 do not involve a change in structures, systems, or components that would create a new or different kind of accident from any accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report.

The proposed change to eliminate the head spray piping and the addition of a bolted flange on the reactor coolant pressure boundary enhances plant safety by eliminating a source of pipe whip and potential leakage. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be tested in conjunction with the containment integrated leak rate test.

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

3. Does not involve a significant reduction in the margin of safety?

Response: No.

The proposed change to delete the head spray valves from Tables 3.3.2–1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 does not reduce any margin of safety as defined in the Technical Specifications or Bases. The bolted flange that will be installed on the head spray penetration will maintain the integrity of the reactor coolant pressure boundary. This flange would then be tested as part of the reactor pressure vessel hydrostatic test. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be tested as part of the containment integrated leak rate test.

Accordingly, based on the above, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Darrell Roberts, Acting.

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

Date of amendment request: October 13, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) limiting conditions for operation 3.8.4, 3.8.5, and 3.8.6, on direct current sources, operating and shutdown, and battery cell parameters. The proposed amendments creates TS 5.5.19, for a battery monitoring and maintenance program. The bases are revised to be consistent with these changes. The proposed amendments are based on Technical Specification Task Force (TSTF) Traveler, TSTF–360, Revision 1.

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

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

No. The proposed changes increase the Completion Time for an inoperable battery, relocate preventative maintenance requirements to licensee controlled programs, and generally restructure the TS [technical specification] requirements for DC [direct current] sources. The revised requirements will allow licensed operators to focus their attention on battery parameters that are indicative of battery operability as opposed to preventative maintenance issues. The increased Completion Time for an inoperable battery will allow corrective maintenance to be accomplished via a more orderly and effective work process. It will also minimize the potential for an additional shutdown/restart transient to comply with the TS in order to accomplish the required maintenance. The DC sources are not initiators to any analyzed accident sequence. Operation in accordance with the proposed TS will continue to ensure that the DC sources remain capable of performing their safety function and that all analyzed accidents will continue to be mitigated as previously analyzed.

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

No. The proposed changes do not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. Plant operation will not be altered, and all safety functions previously addressed in accident analyses will continue to be performed.

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

No. The proposed changes will not adversely affect operation of plant equipment—principally the four Class 1E DC sources and the equipment supported by them. The changes aimed at restructuring the TS requirements for DC sources will have the effect of reducing the burden on licensed operators by focusing the TS requirements on conditions that impair DC source operability. Requirements related to preventive maintenance will be addressed via new Specification 5.5.19 and the plant maintenance program. Margin to the battery operability requirements will continue to be maintained at current levels in accordance with IEEE–450. The extended Completion Time for an inoperable battery has been shown to have a negligible impact on plant risk using the criteria of Regulatory Guides 1.174 and 1.177.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, Nations Bank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: December 15, 2003.

Description of amendment request: The proposed amendments would revise Technical Specifications surveillance requirement (SR) 3.3.1.2 for the nuclear instrumentation system power range daily surveillance when operating above 15-percent rated thermal power. In addition, the format of SR 3.3.1.3 is being revised to be consistent with the format of the proposed change to SR 3.3.1.2.

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

The proposed change to SR [surveillance requirement] 3.3.1.2 does not significantly increase the probability or consequences of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. This modification does not directly initiate an accident. The consequences of accidents previously evaluated in the FSAR are not adversely affected by this proposed change because the change to the NIS [nuclear instrumentation system] Power Range channel adjustment requirement ensures the conservative response of the channel even at part power levels. The proposed change to SR 3.3.1.3 is to change the format consistent with the format of the proposed change to SR 3.3.1.2.

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

The proposed change to SR 3.3.1.2 does not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed Technical Specifications change does not challenge the performance or integrity of any safety-related systems. The proposed change to SR 3.3.1.3 is to change the format to be consistent with the format of the proposed change to SR 3.3.1.2.

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

The proposed change to SR 3.3.1.2 does not involve a significant reduction in a margin of safety. The proposed change does require a revision to the criterion for implementation of Power Range channel adjustment based on secondary power calorimetric calculation; however, the change does not eliminate any RTS [reactor trip system] surveillances or alter the frequency of surveillances required by the Technical Specifications. The revision to the criterion for implementation of the daily surveillance will have a conservative effect on the performance of the NIS Power Range channel, particularly at part power after normalization at 100% RTP [rated thermal power] conditions. The nominal trip setpoints specified by the Technical Specifications and the safety analysis limits assumed in the transient and accident analysis are unchanged. The margin of safety associated with the acceptance criteria for any accident is unchanged. The proposed change to SR 3.3.1.3 is to change the format to be consistent with the format of the proposed change to SR 3.3.1.2.

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

Attorney for licensee: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: December 1, 2003.

Description of amendment request: The proposed amendments would revise Technical Specifications Section 5.5.12, ''Primary Leakage Rate Testing Program,'' to change the peak calculated post accident primary containment internal pressure to support a 10 psi increase in the nominal Unit 1 and 2 reactor operating pressure.

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

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

The proposed change to TS [technical specification] section 5.5.12, ''Primary Containment Leakage Rate Testing Program'', involves an increase to the peak post accident primary containment pressure. It does not involve physical changes to the primary containment structure itself, nor to any of its support systems and components, nor does it involve changes to any other systems and components designed for the prevention of previously analyzed events. Consequently, the proposed amendment does not involve a significant increase in the probability of occurrence of a previously evaluated event.

The increase in operating pressure for the Hatch reactors from 1035 psig to 1045 psig results in an increase to the peak post-accident primary containment internal pressure. This pressure increases from 50.5 to 50.8 psig for Unit 1 and from 46.9 to 47.3 psig for Unit 2. This is a very small increase with respect to the Unit 1 and 2 primary containment design pressure of 56 psig and with the maximum code allowable pressure of 62 psig. The primary containment thus remains capable of withstanding the post accident pressure and thus the consequences of a previously evaluated event are not increased.

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

The primary containment boundary will not be altered by the proposed change to

Technical Specifications sections 5.5.12, Primary Containment Leakage Rate Testing Program. Furthermore, the primary containment will function as presently described in the Updated Final Safety Analysis Report and will be subject to the same structural and functional requirements. The containment will be operated, maintained and surveilled as before, with the exception of the increased peak post accident pressure, which changes the post accident test pressure acceptance criteria. As a result, no new modes of operation are introduced by this Technical Specifications change and therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

The change in the analyzed peak post accident containment pressure will require that the containment be tested to ensure that it meets leakage acceptance criteria at the new pressures of 50.8 psig and 47.3 psig for Units 1 and 2 respectively. Therefore, the primary containment's ability to sustain the slightly higher pressures will be verified during leak rate testing at the required intervals.

The Unit 1 peak pressure increases from 50.5 to 50.8 psig and the Unit 2 pressure increases from 46.9 to 47.3 psig. The primary containment design pressure is 56 psig for both units and the maximum code allowable pressure is 62 psig. Therefore, the margin to the design and maximum code allowable pressures has not been significantly affected. As a result, this proposed Technical Specifications change does not significantly reduce the margin of safety associated with the primary containment function.

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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: John A. Nakoski.

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

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

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

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 12, 2003, as revised by letter dated December 5, 2003.

Brief description of amendment request: By letter dated December 5, 2003, Entergy submitted a revised application for amendment to Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to add a provision to the APPLICABILITY function that will eliminate the requirement that the Residual Heat Removal System Isolation, Reactor Vessel Water Level-Low, Level 3, be OPERABLE under certain conditions during refueling outages. Specifically, the proposed change requested in the original application dated May 12, 2003, would remove the requirement for this isolation function, specified in Table 3.3.6.1–1, when the upper containment reactor cavity is at the High Water Level condition specified in TS 3.5.2, ''Emergency Core Cooling Systems Shutdown.'' The revised application adds a new surveillance requirement (SR) 3.3.6.1.9 to verify that the water level in the upper containment pool is greater than or equal to 22 feet 8 inches above the reactor pressure vessel flange every four hours, and adds a footnote to Table 3.3.6.1–1, Item 5.b, for MODE 5 that states that the function is not required when the upper containment reactor cavity and transfer canal gates are removed and SR 3.3.6.1.9 is met. The proposed SR and footnote are only applicable in MODE 5. The May 12, 2003, application was previously noticed in the **Federal Register** on June 10, 2003 (68 FR 34665).

*Date of publication of individual notice in **Federal Register:*** December 15, 2003 (68 FR 69726).

Expiration date of individual notice: January 14, 2004.

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 email to *pdr@nrc.gov.*

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

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

Date of application for amendments: July 14, 2003, as supplemented by letter dated October 1, 2003.

Brief description of amendments: The amendments extend from 1 hour to 24 hours the completion time for Condition B of Technical Specification 3.5.1, which defines requirements for the restoration of an emergency core cooling system accumulator when it has been declared inoperable for a reason other than boron concentration.

Date of issuance: December 23, 2003.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance December 23, 2003.

Amendment Nos.: 211, 205, 218, and 200.

Renewed Facility Operating License Nos. NPF–35, NPF–52, NPF–9, and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** October 14, 2003 (68 FR 59214).

The supplement dated October 1, 2003, provided clarifying information that did not change the scope of the July 14, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 23, 2003.

No significant hazards consideration comments received: No.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: January 9, 2003.

Brief description of amendment: The proposed Technical Specification (TS) amendment request changes the definition of a Logic System Functional Test, deletes the definition of a Simulated Automatic Actuation, clarifies Surveillance Requirement 4.5.G.1.a regarding simulated automatic actuation testing, and revises associated TS Bases.

Date of Issuance: December 23, 2003.

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

Amendment No.: 216.

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

Date of initial notice in ***Federal Register:*** February 4, 2003 (68 FR 5674).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated December 23, 2003.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: October 24, 2003.

Brief description of amendment: The amendment revises TS 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' for the condition of having one or more SDV vent or drain lines with one valve inoperable.

Date of issuance: December 30, 2003.

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

Amendment No: 161.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 25, 2003 (68 FR 66135).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 2003.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: May 8, 2003, as supplemented by letter dated October 24, 2003.

Brief description of amendment: The amendment changes Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to add a note allowing intermittent opening of penetration flow paths, under administrative control, that are isolated to comply with TS ACTIONS and to revise the operability requirement for the Reactor Core Isolation Cooling (RCIC) steam supply line low pressure isolation instrumentation to be consistent with the RCIC system operability requirements.

Date of issuance: January 8, 2004.

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

Amendment No: 162.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 10, 2003 (68 FR 34664).

The October 24, 2003, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2004.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: September 3, 2003.

Brief description of amendment: The amendment modified Technical Specification (TS) requirements for mode change limitations to adopt the TS Task Force (TSTF) change TSTF–359, ''Increase Flexibility in Mode Restraints.''

Date of issuance: January 5, 2004.

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

Amendment No.: 109.

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

Date of initial notice in ***Federal Register:*** September 30, 2003 (68 FR 56345).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated January 5, 2004.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: July 1, 2003, as supported by letter dated June 16, 2003, and supplemented on November 11, 2003.

Brief description of amendment: The requested changes revise License Condition 2.C.(10) to document changes to the Salem Post-Fire Safe Shutdown (SSD) strategy for Fire Areas 2–FA–AB–64B, 2–FA–AB–84B, and 2–FA–AB–84C. The licensee requested changes to the SSD as a result of recent plant modifications implemented in response to the resolution of Electrical Raceway Fire Barrier System issues at Salem.

Date of issuance: January 7, 2004.

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

Amendment No.: 242.

Facility Operating License No. DPR–75: This amendment revised the Facility Operating License.

Date of initial notice in ***Federal Register:*** July 16, 2003 (68 FR 42134). The supporting and supplemental letters dated June 16, and November 11, 2003, contained clarifying information that did not change the NRC staff's proposed finding of no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 7, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 22, 2003.

Brief description of amendments: The amendments revise Technical Specification (TS) 3/4.3.2, ''Engineering Safety Features Actuation System Instrumentation,'' and TS 3/4.9.9, ''Refueling Operations—Containment Ventilation Isolation System,'' governing radiation monitoring instrumentation, to relax restrictions on containment purge valve operation.

Date of issuance: January 5, 2004.

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

Amendment Nos.: 160 and 150.

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

*Date of initial notice in **Federal Register:*** October 14, 2003 (68 FR 59221).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 5, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 14, 2003, as supplemented by letters dated September 5 and November 7, 2003.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.3.4.1, ''End-Of-Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation,'' and TS 3.7.5, ''Main Turbine Bypass System,'' to reference additional core limits adjustment factors for linear heat generation rate for equipment out-of-service conditions. Also, Section b of TS 5.65. ''Core Operating Limits Report (COLR),'' was revised to add references to the Framatome Advanced Nuclear Power analytical methods what will be used in the upcoming fuel cycles to determine core operating limits.

Date of issuance: December 30, 2003.

Effective date: Date of issuance, to be implemented within 60 days from the completion of Unit 3 Spring 2004 and Unit 2 Spring 2005 refueling outages.

Amendment Nos.: 287 & 245.

Facility Operating License Nos. DPR–52, and DPR–68. Amendments revised the TSs.

*Date of initial notice in **Federal Register:*** May 27, 2003 (68 FR 28858).

TVA's supplemental letters provided clarifying information that did not expand the scope of the original application or change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 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 email to *pdr@nrc.gov.*

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By February 19, 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 27, 2003 as supplemented by letter dated December 27 and two letters dated December 28, 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 (SDG) 22 from 21 days to 113 days as a one-time change for the purpose of making repairs to SDG 22.

Date of issuance: December 30, 2003.

Effective date: December 30, 2003.

Amendment No.: 149.

Facility Operating License No. NPF–80: Amendment revised 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 30, 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 13th day of January 2004.

For the Nuclear Regulatory Commission.

Eric J. Leeds,

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

[FR Doc. 04–1104 Filed 1–16–04; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

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

Extension:

Rule 57(a); SEC File No. 270–376; OMB Control No. 3235–0428.

Form U–57; SEC File No. 270–376; OMB Control No. 3235–0428.

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

Under rule 57(a) a Form U–57 must be used by a person filing under sections 33(a)(3)(B) and 33(c)(1) of the Act. The 101 annual responses together incur about 405 burden hours to comply with these requirements. The Commission estimates that the total annual reporting and recordkeeping burden is 405 (101 annual responses x 10 hours = 1010 burden hours). This represents the same estimated hours annually in the paperwork burden from the prior estimate. The Commission needs the information required by Rule 57(a) in order for the Commission to be informed of when a registered holding company becomes a foreign utility company or when it acquires a foreign utility company. The Commission uses this information to determine the existence of detriment to the interests the Act was designed to protect. Compliance with the requirements to provide the information is mandatory. The information will not be kept confidential.

The estimate of average burden hours is made solely for the purposes of the Paperwork Reduction Act. The estimate is not derived from a comprehensive or even a representative survey or study of the costs of Commission rules and forms.

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: January 7, 2004.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 04–1072 Filed 1–16–04; 8:45 am]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, 450 Fifth Street, NW, Washington, DC 20549.

Extension: Rule 55; SEC File No. 270–376; OMB Control No. 3235–0430.

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

Under rule 55, a filing must be under section 33(c)(1) of the Act for a ''safe harbor'' for acquisitions of foreign utility companies by registered holding companies. The filing is made only for foreign utility companies that meet specific criteria. Rule 55 is a proposal, and has not yet been adopted in final. The Commission estimates that 11 annual responses together incur about 39,710 burden hours to comply with these requirements. The Commission estimates that the total annual reporting and recordkeeping burden is 110 (11 annual responses × 10 hours = 110 burden hours). This represents a decrease of 39,600 hours annually in the paperwork burden from the prior estimate, and this decrease was caused by a decrease in the number of annual responses. The Commission needs the information because it gives the registered holding company a ''safe harbor'' when it acquires a foreign utility company that meets specified criteria. The Commission uses this information to determine the existence of detriment to the interests the Act was designed to protect. Compliance with the requirements to provide the information is mandatory. The information will not be kept confidential.

The estimate of average burden hours is made solely for the purposes of the Paperwork Reduction Act. The estimate is not derived from a comprehensive or even a representative survey or study of the costs of Commission rules and forms.

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.