Surry Power Station, Units 1 and 2
Docket Nos. 50–280 and 50–281
License Nos. DPR–32 and DPR–37

Oconee Nuclear Station, Units 1, 2 and 3
Docket Nos. 50–269, 50–270 and 50–287
License Nos. DPR–38, DPR–47 and DPR–55

H.B. Robinson Steam Electric Plant, Unit 2
Docket No. 50–261
License No. DPR–23

St. Lucie Nuclear Plant, Units 1 and 2
Docket Nos. 50–335 and 50–389
License Nos. DPR–67 and NPF–16

Turkey Point Nuclear Generating Station, Units 3 and 4
Docket Nos. 50–250 and 50–251
License Nos. DPR–31 and DPR–41

Sequoyah Nuclear Plant, Units 1 and 2
Docket Nos. 50–327 and 50–328
License Nos. DPR–77 and DPR–79

Watts Bar Nuclear Plant, Unit 1
Docket No. 50–390
License No. NPF–90

Virgil C. Summer Nuclear Station, Unit 1
Docket No. 50–395
License No. NPF–12

Vogtle Electric Generating Plant, Units 1 and 2
Docket Nos. 50–424 and 50–425
License Nos. NPF–68 and NPF–81

Braidwood Station, Units 1 and 2
Docket Nos. STN 50–456 and STN 50–457
License Nos. NPF–72 and NPF–77

Byron Station, Units 1 and 2
Docket Nos. STN 50–454 and STN 50–455
License Nos. NPF–37 and NPF–66

Donald C. Cook Nuclear Plant, Units 1 and 2
Docket Nos. 50–315 and 50–316
License Nos. DPR–58 and DPR–74

Davis-Besse Nuclear Power Station, Unit 1
Docket No. 50–346
License No. NPF–3

Kewaunee Nuclear Power Plant
Docket No. 50–305
License No. DPR–43

Palisades Plant
Docket No. 50–255
License No. DPR–20

Point Beach Nuclear Plant, Units 1 and 2
Docket Nos. 50–266 and 50–301
License Nos. DPR–24 and DPR–27

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50–282 and 50–306
License Nos. DPR–42 and DPR–60

Arkansas Nuclear One, Units 1 and 2
Docket Nos. 50–313 and 50–368
License Nos. DPR–51 and NPF–6

Callaway Plant, Unit 1
Docket No. 50–483
License No. NPF–30

Comanche Peak Steam Electric Station, Units 1 and 2
Docket Nos. 50–445 and 50–446
License Nos. NPF–87 and NPF–89

Diablo Canyon Nuclear Power Plant, Units 1 and 2
Docket Nos. 50–275 and 50–323
License Nos. DPR–80 and DPR–82

Fort Calhoun Station, Unit 1
Docket No. 50–285
License No. DPR–40

Palo Verde Nuclear Generating Station, Units 1, 2 and 3
Docket Nos. STN 50–528, STN 50–529 and STN 50–530
License Nos. NPF–41, NPF–51 and NPF–74

San Onofre Nuclear Station, Units 2 and 3
Docket Nos. 50–361 and 50–362
License Nos. NPF–10 and NPF–15

South Texas Project Electric Generating Station, Units 1 and 2
Docket Nos. 50–498 and 50–499
License Nos. NPF–76 and NPF–80

Waterford Steam Electric Generating Station, Unit 3
Docket No. 50–382
License No. NPF–38

Wolf Creek Generating Station, Unit 1
Docket No. 50–482
License No. NPF–42

[FR Doc. 03–3835 Filed 2–14–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATE: Weeks of February 17, 24, March 3, 10, 17, 24, 2003.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of February 17, 2003

There are no meetings scheduled for the Week of February 17, 2003.

Week of February 24, 2003—Tentative

There are no meetings scheduled for the Week of February 24, 2003.

Week of March 3, 2003—Tentative

Monday, March 3, 2003

10 a.m.—Briefing on Status of Office of Nuclear Material Safety and Safeguards (NMSS) Programs—Waste Safety (Public Meeting) (Contact: Claudia Seelig, 301–415–7243)

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

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

Week of March 10, 2003—Tentative

There are no meetings scheduled for the Week of March 10, 2003.

Week of March 17, 2003—Tentative

Thursday, March 20, 2003

10 a.m.—Briefing on Status of Office of Nuclear Security and Incident Response (NSIR) Programs, Performance, and Plans (Closed—Ex. 1)

2 p.m.—Discussion of Management Issues (Closed—Ex. 2)

Week of March 24, 2003—Tentative

Thursday, March 27, 2003

2 p.m.—Briefing on Status of Office of Research (RES) Programs, Performance, and Plans (Public Meeting)

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

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

* * * * *

Additional Information: "Meeting with National Association of Regulatory Utility Commissioners (NARUC)," originally scheduled for February 24, 2003, has been canceled.

* * * * *

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

* * * * *

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

Dated: February 12, 2003.

R. Michelle Schroll,

Acting Technical Coordinator, Office of the Secretary.

[FR Doc. 03–3934 Filed 2–13–03; 11:19 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, January 24, 2003, through February 6, 2003. The last biweekly notice was published on February 4, 2003 (68 FR 5668).

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 March 20, 2003, 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,[1] 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

[1] The most recent version of title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 CFR 2.714(d), please *see* 67 FR 20884; April 29, 2002.

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, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendments request: December 13, 2002.

Description of amendments request: The proposed amendments would revise Technical Specification 3.5.2, Emergency Core Cooling System—Operating, by removing the Note that modifies the Limiting Condition for Operation. The proposed change would remove the requirement to have the charging pumps operable when thermal power is greater than 80% of rated thermal power (RTP). The proposed change would also remove Surveillance Requirement 3.5.2.4 for verifying the required charging pump flow rate.

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 charging pumps were credited in the previous analysis to mitigate the consequences of a small-break loss-of-coolant accident (LOCA) above 80% of rated thermal power (RTP). The charging pumps were not considered to be an initiator of the accident. The new analysis for the small-break LOCA does not assume the charging pumps are initiators of the accident. Therefore, removing the requirement to maintain the charging pumps operable above 80% RTP and removing Surveillance Requirement 3.5.2.4 from the Technical Specification does not involve a significant increase in the probability of an accident previously evaluated.

The consequence of a small-break LOCA is the potential for inadequate core cooling and decreased negative reactivity such that the reactor core is not protected after the design basis event. The previous analysis for the small-break LOCA above 80% RTP assumed unborated flow from a single charging pump to ensure there was adequate cooling flow delivered to the Reactor Coolant System. The revised small-break LOCA analysis was performed such that flow from the charging pumps was not credited. Since the charging pump flow is no longer credited in the small-break LOCA analysis, the proposed changes do not involve a significant increase in the consequences of a small-break LOCA.

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

2. Would not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

This request[ed] change does not involve a change in the operation of the plant and no new accident initiation mechanism is created by the proposed changes. Since the charging pump flow is no longer credited in the small-break LOCA analysis, the requirement to have the charging pumps operable above 80% RTP and the charging pump Surveillance Requirement 3.5.2.4 can be removed from the Technical Specification. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

The safety function of the Emergency Core Cooling System is to provide core cooling and negative reactivity, to ensure that the reactor core is protected after design basis events. For a small-break LOCA, the previous analysis credited flow from the charging pumps above 80% RTP to supply supplemental cooling flow to the Reactor Coolant System. Credit for flow from a single charging pump was only taken for the water inventory.

The revised small-break LOCA analysis was performed using the newest Nuclear Regulatory Commission accepted versions of the Westinghouse evaluation models for Combustion Engineering designed pressurized water reactors. The revised small-break LOCA analysis incorporated several changes to plant parameters used in the analysis, one of which was the elimination of the need to credit the charging pump flow above 80% RTP. Since the charging pump flow is no longer credited in the small-break LOCA analysis, the requirement to have the charging pumps operable above 80% RTP and charging pump Surveillance Requirement 3.5.2.4 can be removed from the Technical Specification.

The proposed change to Technical Specification 3.5.2 does not modify any other charging pump requirements specified in the Technical Requirements Manual (*e.g.*, requirements on charging pump availability for boration and cooldown remain in effect).

Therefore, the safety function of the Emergency Core Cooling System is maintained and the margin of safety is not significantly reduced by the proposed changes.

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

NRC Section Chief: Richard J. Laufer.

Carolina Power & Light Company, Docket Nos. 50–325 and 50–324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: November 12, 2002.

Description of amendments request: The proposed amendments would revise the Technical Specifications, as necessary, to support an expansion of the core flow operating range (*i.e.*, Maximum Extended Load Line Limit Analysis Plus (MELLLA+)). As part of the MELLLA+ implementation, Carolina Power & Light Company would implement the Detect and Suppress Solution-Confirmation Density (DSS–CD) approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities.

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:

10 CFR 50.91(a) states ''At the time a licensee requests an amendment, it must provide to the Commission its analysis about the issue of no significant hazards consideration using the standards in § 50.92.'' The following provides this analysis for the MELLLA+ operating range to a minimum core flow rate of 85% of rated with 120% of the original licensed thermal power.

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

The expansion of the core operating range discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of a DBA [design-basis accident] 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 (ASME [American Society of Mechanical Engineers] code, IEEE [Institute of Electrical and Electronics Engineers] standards, NEMA [National Electrical Manufacturers Association] standards, Regulatory Guides, etc.). An evaluation of the probabilistic safety assessments concludes that the calculated core damage frequencies do not significantly change due to the MELLLA+ operating range expansion. 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 challenge to safety related equipment results from the MELLLA+ operating range expansion. The changes in consequences of hypothetical accidents, which occur from operation in the MELLLA+ region, are in all cases insignificant. The MELLLA+ accident evaluations do not exceed any NRC-approved acceptance limits. The spectrum of hypothetical accidents and abnormal operational occurrences has been investigated, and will 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 Safety Limit Minimum Critical Power Ratio (SLMCPR) are met, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in [GE Nuclear Energy, ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A and NEDE–24011–P–A–US, (latest approved revision)]. Challenges to fuel (ECCS [emergency core cooling system] performance) are evaluated, and shown to still meet the criteria of 10 CFR 50.46, Appendix K, Regulatory Guide 1.70, and UFSAR [Updated Final Safety Analysis Report] Section 6.3. Challenges to the containment have been evaluated, and the containment and its associated cooling systems meet 10 CFR 50 Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment. Radiological release events (accidents) have been evaluated, and shown to meet the regulatory limits of 10 CFR 50.67.

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

The MELLLA+ operating range expansion will not create the possibility of a new or different kind of accident from any accident previously evaluated. 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 UFSAR, has been evaluated, and no new or different kind of accident has been identified. The MELLLA+ operating range expansion uses fully developed technology, and applies it within the capabilities of existing plant equipment. The technology includes NRC approved codes, standards and methods applied in accordance with existing regulatory criteria.

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

The MELLLA+ operating range expansion will not involve a significant reduction in a margin of safety. 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 criterion is 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.

Conclusion: A MELLLA+ operating range expansion to a minimum core flow rate of 85% of rated with 120% of original licensed thermal power has been investigated. The BSEP [Brunswick Steam Electric Plant] licensing requirements have been evaluated and it has been demonstrated that this MELLLA+ operating range expansion can be accommodated:

- Without a significant increase in the probability or consequences of an accident previously evaluated,
- Without creating the possibility of a new or different kind of accident from any accident previously evaluated, and
- Without exceeding any presently existing regulatory limits or acceptance criteria applicable to the plant, which might cause a reduction in a margin of safety.

Having made negative declarations regarding the 10 CFR 50.92 criteria, this assessment concludes that an operating range expansion to a minimum core flow rate of 85% of rated with 120% of original licensed thermal power does not involve a Significant Hazards Consideration.

10 CFR 50.91(a) states ''At the time a licensee requests an amendment, it must provide to the Commission its analysis about the issue of no significant hazards consideration using the standards in § 50.92.'' The following provides this analysis for the DSS–CD long-term stability solution.

(1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change will implement DSS–CD as the long-term stability solution. The DSS–CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion 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 General Design Criteria (GDC) 10 and 12 of 10 CFR part 50, Appendix A is accomplished via an automatic action. The DSS–CD introduces an enhanced detection algorithm that detects the inception of power oscillations and generates an earlier power suppression trip signal exclusively based on successive period confirmation recognition. The existing Option III algorithms are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

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. In addition, the DSS–CD solution defense-in-depth features incorporate all the backup scram algorithms plus the licensed scram feature of the existing Option III system. 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.

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. As stated above, the DSS–CD solution meets the requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel SLMCPR.

Based on the above, the operation of the DSS–CD solution within the framework of the Option III OPRM hardware will not increase the probability or consequences of an accident previously evaluated.

(2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The DSS–CD solution operates within the existing Option III OPRM [Oscillation Power Range Monitor] hardware. 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 hardware design 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.

Based on the above, the proposed change to the DSS–CD solution will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Will the change involve a significant reduction in a margin of safety?

The DSS–CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion to terminate 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. As a result, there is no impact on the MCPR [minimum critical power ratio] Safety Limit identified for an instability event.

The current Option III algorithms (Period Based Detection, Amplitude Based, and Growth Rate) are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

Based on the above, the proposed change will not involve a significant reduction in the margin of safety.

Conclusions: The DSS–CD stability solution has been investigated. The BSEP licensing requirements have been evaluated and it has been demonstrated that the DSS–CD stability solution can be accommodated:

• Without a significant increase in the probability or consequences of an accident previously evaluated,

• Without creating the possibility of a new or different kind of accident from any accident previously evaluated, and

• Without exceeding any presently existing regulatory limits or acceptance criteria applicable to the plant, which might cause a reduction in a margin of safety.

Having made negative declarations regarding the 10 CFR 50.92 criteria, this assessment concludes that the DSS–CD stability solution does not involve a Significant Hazards Consideration.

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

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602. *NRC Section Chief:* Allen G. Howe.

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: June 7, 2002, supplemented by letter dated January 8, 2003.

Description of amendment request: The proposed amendments would revise the Updated Final Safety Analysis Report to eliminate credit for the flow path from the spent fuel pool to the high pressure injection pump as one source of primary system makeup following a tornado. The proposed amendments would also credit the Standby Shutdown Facility as the assured means of achieving safe shutdown for all three Oconee units following a tornado. By letter dated January 8, 2003, Duke Energy Corporation provided a revised No Significant Hazards Consideration (NSHC) that supercedes the NSHC that was noticed in the **Federal Register** on July 23, 2002 (67 FR 48216).

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:

Pursuant to 10 CFR 50.91, Duke Energy Corporation (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures 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.

The changes being requested in this amendment request involve (1) the elimination of the Spent Fuel Pool [SFP] as a suction source to a High Pressure Injection [HPI] pump for primary system make-up, and (2) to fully credit the Standby Shutdown Facility (SSF) as the primary assured means of achieving safe shutdown of all three units following a tornado. Following the modification to fully tornado protect the SSF, this facility becomes the station's assured flow path for both primary make-up and secondary decay heat removal for all three units.

Although the probability of a severe tornado strike at the station does not change, new tornado insights gained from a review of the current external event risk analysis have resulted in an enhanced risk model that more accurately characterizes station tornado damage risk. The proposed changes are part of the revised tornado mitigation strategy that provides for an assured, deterministic success path rather than the current strategy that is based on risk insights and diversity for achieving safe shutdown. This effort has resulted in an overall reduction in tornado risk at the station and consequently, would not result in a significant increase in the consequences of an accident previously evaluated.

Other than the fortification of walls of existing structures to harden them against tornado damage, there are no physical changes to the plant structures, systems, or components (SSCs), nor are there any changes to safety limits or set points. Also, no new radiological release pathways are created.

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

The changes being proposed in this amendment request do not create the possibility of a new or different kind of accident from any accident previously evaluated. The initial placement of the SFP–HPI flow path into the LB [licensing basis] was based on 1989 risk analyses that showed a potential need for primary make-up due to inventory losses from a reactor coolant pump (RCP) seal loss-of-cooling accident (LOCA). The upgrade of the RCP seals has significantly reduced the probability of a seal LOCA and subsequently, alleviated the initial reliance on the SFP–HPI flow path for primary make-up. If multi-unit primary make-up and decay heat removal are required following an event, the tornado protected SSF RBMU [sic] [(RCMU) reactor coolant makeup] or SSF ASW [auxiliary service water] pumps have the capabilities to perform these functions for all three units.

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

As mentioned previously, new tornado insights gained from a review of the current external event risk analysis have resulted in an enhanced risk model that more accurately characterizes station tornado damage risk. The proposed changes are part of the revised tornado mitigation strategy that provides for an assured, deterministic success path rather than a strategy that is based on risk insights and diversity for achieving safe shutdown.

There is no safety limit, set point, or design parameter changes required. The integrity of the fuel cladding, reactor coolant system, and containment are preserved. Thus, 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: December 30, 2002.

Description of amendment request: The proposed amendment revises two technical specifications (TSs). The first change proposes to revise TS 2.1.1.2, ''Minimum Critical Power Ratio Safety Limit (MCPRSL)'' to support operation during Cycle 17 with a mixed core. The second change proposes to revise the local power range monitor (LPRM) calibration frequency specified in the TS for the oscillation power range monitor (OPRM) in Surveillance Requirement (SR) 3.3.1.3.2. This change will correct an inconsistency between the LPRM calibration frequency specified in SR 3.3.1.3.2 and SR 3.3.1.1.7, ''Reactor Protection System (RPS) Instrumentation.''

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

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

1. The requested change to TS 2.1.1.2, MCPRSL to support the cycle 17 core loading does not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigation systems, and do not introduce any new accident initiation mechanisms. The consequences of accidents previously evaluated are not changed because the number of rods that are protected from transition boiling is predicted to be greater than 99.9 percent which meets the acceptance criterion in NUREG–0800, Section 4.4.

2. The requested change to SR 3.3.1.3.2, OPRM/LPRM calibration frequency, does not involve a modification to the plant or introduce the probability of an operator error. The LPRMs are not the precursor to any accident. Making the LPRM surveillance frequency for the OPRM consistent with that approved for the RPS/APRM [reactor protection system/average power range monitor] does not change system reliability. The proposed LPRM surveillance frequency is supported by the uncertainties used to perform the MCPRSL analyses. Therefore, the number of rods that are calculated to experience transition boiling during normal operation or anticipated operational occurrences will not be changed and the consequences of these events will not be increased.

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

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

1. The ATRIUM–10 fuel to be used in cycle 17 is compatible with the co-resident SVEA–96 fuel. This compatibility is demonstrated by application of the FRA–ANP critical power methodology to the core design that includes the ATRIUM–10 and SVEA–96 fuel. The proposed changes do not represent any new modes of operation, changes in setpoints or plant modifications other than those required for the reactor core. The change does not introduce new postulated accident precursors or mitigation systems. Reload design and analysis will be performed in accordance with approved NRC methodology.

2. Increasing the time interval for the OPRM/LPRM surveillance reduces the frequency to be consistent with the LPRM surveillance frequency for the RPS/APRM and does not involve a modification to the plant, introduce a new operator error or revise setpoints.

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

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

1. The proposed MCPRSL does not involve a significant reduction in the margin of safety associated with the criterion set forth in NUREG–0800, section 4.4. The safety limit established for the core ensures that the criterion for the number of fuel rods allowed to experience transition boiling will be maintained for normal plant operation and anticipated operational transients.

The core operating limits will continue to be determined using methodologies that have been approved by the NRC.

2. The proposed LPRM surveillance frequency is supported by the uncertainties used to perform the MCPRSL analyses. Therefore, the number of rods that are calculated to experience transition boiling during normal operation or anticipated operational occurrences will not be changed.

Therefore, implementation of the change to the MCPRSL and the LPRM surveillance frequency does not involve a significant reduction in [a] margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: October 10, 2002, as supplemented on November 22, 2002, and January 28, 2003. This notice supercedes 67 FR 68735 published on November 12, 2002, which erroneously stated that the October 10, 2002, application was a supplement of the licensee's application dated December 12, 2001. The October 10, 2002, replaced the December 12, 2001, application. This notice also adds supplements dated November 22, 2002, and January 28, 2003.

Description of amendment request: The proposed amendment would change the Technical Specification Tables 3.2.A, 3.2.B, 4.2.A, and 4.2.B. The proposed changes affect various instrument trip level settings and decreases the calibration frequencies for a variety of instruments. The proposed changes also involve clarifications to the Reactor Water Cleanup system trip configuration and the titles of certain trip systems. In addition, the proposed changes would make certain editorial and administrative corrections. The proposed setpoint changes and calibration frequencies are based on the licensee's evaluation.

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

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

The methodology used to determine the proposed trip level settings and surveillance intervals ensure adequate performance of the affected instrumentation. In addition, the affected instruments are not initiators of any accident previously evaluated. Therefore, the proposed trip level setting and surveillance intervals will not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes to trip level settings and surveillance intervals were establish using methodologies subject to 10 CFR Appendix B Quality Assurance program and ensure existing radiological limits are met. Therefore, the proposed trip level settings and surveillance intervals will not involve a significant increase in the consequences of an accident previously evaluated.

Other changes are editorial or administrative in nature and can not significantly increase the probability or consequences of an accident previously evaluated.

2. Will not create the possibility of a new or different kind of accident from any accident previously evaluated.

No new or different [kind] of accidents or malfunctions than those previously analyzed in Pilgrim's UFSAR [Updated Final Safety Analysis Report] are introduced by this proposed change because there are no new failure modes introduced. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to trip level settings and surveillance intervals were established using approved methodologies subject to a 10 CFR, Appendix B, Quality Assurance program and existing radiological limits are met. These changes do not impact Pilgrim's configuration or operation.

Editorial and administrative type changes do not impact the operation or configuration of Pilgrim. For the above reasons the proposed change does not result in 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: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: James W. Clifford.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: December 4, 2002. This notice supercedes 68 FR 2801 published on January 21, 2003, which erroneously stated that the December 4, 2002, application was a supplement of the licensee's application dated May 1, 2002. The December 4, 2002, application replaced the May 1, 2002, application in its entirety.

Description of amendment request: The proposed amendment would extend the applicability of the current Pilgrim Nuclear Power Station (Pilgrim) reactor pressure vessel pressure-temperature (P–T) curves through the end of Operating Cycle (OC) 16. The current P–T curves were approved for use in License Amendment 190, dated April 13, 2001, and are limited to use through the end of OC 14. The proposed change would delete the 20 and 32 Effective Full Power Year (EFPY) curves and replace the wording of the title blocks to allow use through the end of OC 16. The proposed amendment would change Pilgrim Technical Specification Figures 3.6.1, 3.6.2, and 3.6.3.

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

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

The proposed change involves a request to extend the use of the current reactor pressure vessel P–T curves for two additional OCs. The P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR part 50, Appendix G, and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), section XI, Appendix G and Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, and were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor pressure vessel beltline materials. There are no physical changes to the plant or new modes of operation being introduced by the proposed change. Further, the proposed change does not involve a change to any activities or equipment and is not assumed in the safety analysis to initiate any accident sequence. The proposed change does not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the containment of radioactive materials is affected. Additionally, the proposed change will not create any failure mode not bounded by previously evaluated accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The current P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR part 50, Appendix G, and ASME Code, section XI, Appendix G, and were approved by the U.S. Nuclear Regulatory Commission for use through OC 14. The proposed change would extend use of the P–T curves for two additional OCs. No new modes of operation are introduced by the proposed change. Plant operation in compliance with the current P–T curves ensures conditions in which brittle fracture of primary coolant pressure boundary materials is avoided. Accidents involving a breach of the primary coolant pressure boundary have previously been evaluated and no other types of accidents associated with the proposed change have been identified. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor pressure vessel beltline materials and are estimated for 48 effective full-power years. The current curves are approved for use through the end of OC 14 (~19 EFPYs) which provides a conservatism factor of 1.7 between the actual EFPYs at the end of OC 14 and the end-of-life curve (32 EFPY). The change would extend the use of the proposed curves to the end of OC 16 (~23 EFPYs) which provides a conservatism factor of approximately 2.0. The actual EFPYs at the end of OC 16 is bounded by the 48 EFPYs estimated for the current curves. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts 02360–5599.

NRC Section Chief: James W. Clifford.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: December 12, 2002.

Description of amendment request: The proposed amendment would add a new Surveillance Requirement (SR) to the technical specification (TS) section 3.7.5, ''Auxiliary Feedwater (AF) System,'' which requires operation of the diesel-driven AF pump on a monthly frequency (*i.e.*, once every 31 days) for greater than or equal to 15 minutes. The current TS SR 3.7.5.3 requires both the diesel-driven AF pump and the motor-driven AF pump to be operated once per quarter in accordance with the Inservice Testing Program; however, based on operating experience, Braidwood and Byron Stations conduct the diesel-driven AF pump surveillance on a monthly frequency to maintain a high level of assurance that the diesel engine would automatically start when called upon to perform its design basis function.

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

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

The proposed change adds a new TS SR to the AF System TS section 3.7.5. The new SR requires that the diesel-driven AF pump be operated for greater than or equal to 15 minutes every month. Operating experience has shown that conducting the diesel-driven AF pump surveillance on a monthly frequency maintains a high level of assurance that the diesel engine will automatically start when called upon to perform its design basis function.

The previously analyzed events are initiated by the failure of plant structures, systems, or components. The AF system is not considered an initiator for any of these previously analyzed events. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The initial conditions of design basis accident and transient analyses in the Byron/Braidwood Stations Updated Final Safety Analysis Report assume the AF system is operable. The operability of the AF system is assured by the proposed TS SR and is consistent with the initial assumptions of the accident analyses. Since functionality of the diesel engine can be better assured when the diesel-driven AF pump is operated monthly vice quarterly, Exelon is proposing to add a TS SR to operate the diesel-driven AF pump on a monthly frequency. The proposed SR will provide higher confidence that the diesel-driven AF pump will reliably start automatically during an emergency condition, consistent with the AF System design requirements, and continue to mitigate the consequences of the associated design basis accidents. Based on this evaluation, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve the use or installation of new equipment and the 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 changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases. The current diesel-driven AF pump surveillance procedure is already conducted on a monthly basis and has been reviewed, approved and judged appropriate to provide high confidence that the AF diesel engine and pump will reliably start and operate during an emergency condition. The new SR formalizes this monthly surveillance practice in the TS. 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 TS change does not involve a significant reduction in a margin of safety.

The proposed change does not alter any existing setpoints at which protective actions are initiated and no new setpoints or protective actions are introduced. The design and operation of the AF system remains unchanged and maintains the existing margins of safety. Since the increased frequency of the diesel-driven AF pump surveillance test maintains high assurance that the pump's diesel engine will successfully auto-start during an emergency, the proposed additional SR will provide high confidence that the AF system will continue to function as designed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

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.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 23, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) section 6, Administrative Controls, to: (1) relocate administrative requirements discussed in Administrative Letter 95–06 (AL 95–06), ''Relocation of Technical Specification Administrative Controls Related to Quality Assurance,'' to the Operational Quality Assurance Program, (2) change the title of the senior onsite official, and (3) bring the TSs into consistency with changes in 10 CFR part 20.

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

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

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

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

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

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

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

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

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

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

The proposed changes are administrative changes to the Seabrook Station Technical Specifications. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications are not revised nor is the plant design or its method of operation revised by the proposed changes. Thus, it is concluded that these proposed revisions to the Seabrook Station TS do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: October 17, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification 3.7.9, ''Control Room Emergency Filtration System (CREFS),'' by deleting the one-time extension to the allowed outage time (AOT) for CREFS and the exception to the requirements of limiting condition for operation 3.0.4 and surveillance requirement 3.0.4 that were allowed during the AOT.

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 Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The operability of CREFS ensures that the control room will remain habitable for operators during and following all credible accident conditions. The inoperability or failure of CREFS is not an accident initiator or precursor. Therefore, the probability of an accident previously evaluated will not be significantly increased as a result of the proposed change. Because design limitations continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. The evaluation of the effects of the proposed changes indicate that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of any new or different accident nor create any new failure mechanisms.

Equipment important to safety will continue to operate as designed.

Additionally, the changes do not result in any event previously deemed incredible being made credible. The changes also do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components (SSCs) important to safety. Therefore, deleting the one-time extension to the CREFS AOT will not result in 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: John H. O'Neill, Jr., Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: June 28, 2002, as supplemented on December 18, 2002, and January 18, 2003.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TSs) by relaxing the secondary containment requirements and eliminating the Filtration, Ventilation, and Recirculation System (FRVS) charcoal filters.

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

(1) Does the change involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The definition of CORE ALTERATIONS has been revised to define that control rod movement, provided there are no fuel assemblies in the associated core cell, is not a core alteration. This is consistent with Standard Technical Specifications (STS) NUREG–1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4 [Boiling Water Reactor, Type 4].

The TS presently provide a period of 7 days to restore an inoperable FRVS ventilation unit when performing activities with the potential for draining the reactor vessel or discontinue such activities.

Operation of the redundant train will ensure that the remaining subsystem is operable, that no failures, which could prevent automatic actuation, have occurred and that any other failures will be readily detected. This is consistent with STS, NUREG–1433 Vol.1, Rev. 2, Standard Technical Specifications, General Electric Plants, BWR/4.

The proposed changes associated with the FHA [fuel-handling accident] do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report (UFSAR). The FHA for the HCGS [Hope Creek Generating Station] is defined as a drop of a fuel assembly over irradiated assemblies in the reactor core 24 hours after reactor shutdown. AST [accident source term] is used to evaluate the dose consequences of a postulated accident. The FHA has been analyzed without credit for Secondary Containment, Filtration Recirculation and Ventilation System (FRVS), and Control Room Emergency Filtration (CREF) system. The resultant radiological consequences are within the acceptance criteria set forth in 10 CFR 50.67 and Regulatory Guide 1.183. This amendment does not alter the methodology or equipment used directly in fuel handling operations. The equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator. Actual fuel handling operations are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

The Loss of Coolant Accident (LOCA) Dose Calculation has been revised to (1) eliminate credit for the FRVS recirculation charcoal filters, (2) reduce credited efficiency of FRVS vent charcoal filters, (3) reduce Engineered Safety Feature (ESF) leakage from 10 gpm to 1 gpm and (4) reduce control room unfiltered in-leakage to 350 cfm [cubic feet per minute]. These proposed changes do not eliminate any safety system. The changes are only associated with the credit provided by the system in reducing the radiological consequences and therefore, do not affect any accident initiator. The results of that analysis show that the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses are of the same order of magnitude as the previous analysis and remain within the acceptance criteria in 10 CFR 50.67 and Regulatory Guide 1.183.

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

(2) Does the change create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No.

The proposed amendment will not create the possibility for a new or different type of accident from any accident previously evaluated. Changes to the allowable activity in the primary and secondary systems do not result in changes to the design or operation of these systems. The evaluation of the effects of the proposed changes indicates that all design standard and applicable safety criteria limits are met.

Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems.

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

(3) Does the change involve a significant reduction in [a] margin of safety?

Response: No.

The proposed changes revise the TS to establish operational conditions where specific activities represent situations during which significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the regulatory guidelines. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed TS continue to ensure that the TEDE [total effective dose equivalent] for the CR, the EAB, and LPZ are below the corresponding acceptance criteria specified in 10 CFR 50.67 and RG1.183.

Therefore, these changes do not involve a significant reduction in [a] margin of safety.

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

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

NRC Section Chief: James Clifford.

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

Date of amendment request: October 9, 2002, as supplemented November 22, 2002, and December 6, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 6.8.4.f, "Primary Containment Leakage Rate Testing Program," to allow a one-time interval extension to the requirement.

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

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

Response: No.

The proposed revision to section 6.8.4.f adds a one-time extension to the current interval for containment integrated leak rate test (ILRT). The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last ILRT. The proposed extension to ILRT testing cannot increase the probability of an accident previously evaluated since the containment ILRT testing extension is not a modification to plant systems, nor a change to plant operation that could initiate an accident. The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493, "Performance-Based Containment Leak-Test Program," found that very few potential containment leakage paths fail to be identified by Type B and C tests. The NUREG concluded that reducing the ILRT testing frequency to once per twenty years would lead to an imperceptible increase in risk. Containment performance monitoring is performed in accordance with the Maintenance Rule (10 CFR 50.65) and inspections required by American Society of Mechanical Engineers (ASME) code are performed in order to identify indications of containment degradation that could affect leak tightness. Type B and C testing required by the technical specifications (TS) will identify any containment opening, such as valves, that would otherwise be detected by the ILRT. Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. It also recommends the use of risk analysis techniques to ensure and show that the proposed change is consistent with the defense-in-depth philosophy. The increase in large early release frequency (LERF) resulting from a change in the ILRT test frequency from the current once in every 10 years to once in every 15 years is less than 1E–7 per year, thereby meeting Regulatory Guide 1.174 definition of a very small change in risk. The change in conditional containment failure probability (CCFP) is estimated to be 0.25% for the proposed change. These factors show that an ILRT test extension will not represent a significant increase in the consequences of an accident.

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

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

Response: No.

The proposed revision to section 6.8.4.f adds a one-time exception to the current interval for the ILRT. The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last Type A test. Primary containment is designed to contain energy and fission products during and after an event. The Individual Plant

Examination (IPE) identifies events that lead to containment failure. Revision to the ILRT test interval does not change this list of events. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating a new or different kind of accident.

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

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

Response: No.

The proposed revision to section 6.8.4.f adds a one-time extension to the current interval for the ILRT. The current test interval of 10 years, based upon past performance, would be extended on a one-time basis to 15 years from the last ILRT. The proposed extension to ILRT testing interval will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year exception in ILRT leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that the containment leakage rate contributes a very small amount to the individual risk, and that the decrease in Type A testing frequency would have a minimal affect on this risk since most potential leakage paths are detected by Type C testing. Type B and Type C testing will continue to be performed at a frequency currently required by the Technical Specifications (TS). The containment inspections being performed in accordance with ASME, section XI, and Maintenance Rule (10 CFR 50.65) provide a high degree of assurance that the containment will not degrade in a manner that is only detectable by Type A testing.

Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. It also recommends the use of risk analysis techniques to ensure and show that the proposed change is consistent with the defense-in-depth philosophy. The increase in large early release fraction (LERF) resulting from a change in the ILRT test frequency from the current once in every 10 years to once in every 15 years is less than 1E–7 per year, thereby meeting Regulatory Guide 1.174 definition of a very small change in risk. The change in conditional containment failure probability (CCFP) is estimated to be 0.25% for the proposed change.

Therefore, these changes do not involve a significant reduction in 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, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James Clifford.

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

Date of amendment request: July 25, 2002, as supplemented on October 21, 2002.

Description of amendment request: The proposed change would revise Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specifications (TSs) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from the current limit of up to 24 hours, to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the following requirement would be added to SR 4.0.3: ''A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.'' PSEG is also proposing changes to adopt a TS Bases Control Program and changes to SR 4.0.1.

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process (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 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the model NSHC determination for amendments concerning missed surveillances in its original application dated July 25, 2002. The proposed amendment would also make administrative changes to SRs 4.0.1 and 4.0.3 to be consistent with NUREG–1431, Revision 2, ''Standard Technical Specifications, Westinghouse Plants.'' These changes are necessary to make the current Salem TSs compatible with the proposed CLIIP changes for missed surveillances.

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

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

[Specification 4.0.3]

The proposed change relaxes the time allowed to perform a missed Surveillance. The time between Surveillances is not an initiator to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be OPERABLE and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected.

[Specification 4.0.1]

The proposed additional requirement equating failure to meet a surveillance with failure to meet the [limiting condition for operation] is consistent with current interpretation of the technical specifications. This change, along with relocation and rewording of existing requirements from Specification 4.0.3, are administrative in nature and do not adversely affect accident initiators, design functions, facility configuration or the manner of operation or control. The ability of structures, systems and components to perform their intended function remains unaffected.

[Bases Control Program]

The proposed change to adopt a Technical Specification Bases Control Program is also administrative in nature and does not adversely affect accident initiators, design functions, facility configuration or the manner of operation or control. The ability of structures, systems or components to perform their intended function remains unaffected. Future changes to the TS Bases will continue to be administratively controlled in accordance with the requirements of 10 CFR 50.59.

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

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

None of the three proposed changes involves a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

[Specification 4.0.3]

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

[Specification 4.0.1]

The proposed changes to TS 4.0.1, including relocation and rewording of existing requirements from Specification 4.0.3, are administrative in nature and do not reduce the level of programmatic or procedural controls associated with the Surveillance Requirements. There are no substantive differences in meaning or intent between the existing specifications and the corresponding STS requirements. Further, these changes have no impact on equipment design, configuration, analytical basis, setpoints or operation.

[Bases Control Program]

The proposed change to adopt a Technical Specification Bases Control Program is also administrative in nature and does not reduce the level of programmatic or procedural controls associated with the Bases. There is no impact on equipment design, configuration, analytical basis, setpoints or operation.

Thus, there is confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

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, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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 19, 2002.

Description of amendment request: The proposed amendments would change the Operating Licenses and Technical Specifications associated with an increase in the licensed reactor power level of 1.5 percent for each reactor (from 2763 megawatts thermal (MWt) to 2804 MWt).

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:

[Southern Nuclear Company] SNC's conclusion that the proposed change to the Plant Hatch Unit 1 and 2 Operating Licenses and Technical Specifications does not involve a significant hazards consideration is based upon the following:

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

The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of all components and systems that could be affected by this change. Performance requirements for these systems were evaluated and found acceptable. Furthermore, evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. The systems will function as designed. The performance requirements for these systems were evaluated and found acceptable.

The primary loop components (*e.g.*, reactor vessel, reactor internals, control rod drive housings, piping and supports, and recirculation pumps) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, the probability of a structural failure of these components is not increased as a result of this change.

The Nuclear Steam Supply System (NSSS) systems will still perform their intended design functions during normal and accident conditions. The balance-of-plant (BOP) systems and components will continue to meet their applicable structural limits and perform their intended design functions. Thus, the probability of a structural failure of these components is not increased as a result of this change.

The NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves still meet design sizing requirements at the uprated power level.

Because the integrity of the plant will not be affected by operation at the uprated condition, SNC concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the core thermal power uncertainty measurement allows most of the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 2804 MWt. Other analyses performed at a nominal power level were either evaluated or reperformed for the 1.5% increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.5% uprate conditions. Thus, all Plant Hatch Final Safety Analysis Report accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy release remain valid. The source terms used to assess radiological consequences were reviewed and determined to bound operation at the 1.5% uprated condition. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed change will have no adverse effect on any safety-related system or component and does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers confirm that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations were performed using methods that were either reviewed and approved by the NRC, or are in compliance with regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

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: November 14, 2002.

Description of amendment request: The proposed amendment would delete the turbine missile design basis from the Updated Final Safety Analysis Report (UFSAR).

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

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

Response: No.

The turbine missile generation probability will not be significantly increased by elimination of the regulatory commitments in the UFSAR. No plant changes are proposed that would significantly increase the probability of turbine missile generation. Turbine missile generation does not pose a credible threat to safety related components and consequently has no potential to increase radiological consequences.

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

Response: No.

The proposed changes involve no physical modification of the plant or different operating configurations.

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

Response: No.

Turbine missiles do not constitute a credible threat to nuclear safety at STP [South Texas Project]. They are not a consideration in any plant safety analysis. Changing the regulatory commitment with regard to design for turbine missiles has no effect on any margin of safety.

Based upon the analysis provided herein, the proposed amendments do not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 14, 2003 (TS 02–08).

Description of amendment request: The proposed amendment would revise applicability requirements for TS 3.3.9.4, ''Containment Building Penetrations.'' This revision will modify the current applicability requirement associated with movement of ''irradiated fuel'' by adding a new applicability statement for the containment building equipment door.

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 revises the applicability of the containment building penetration function and associated action. This change does not alter the function of the penetrations but does revise when the feature is required to be available for the mitigation of postulated accidents. These penetrations only function to minimize the release of radioactive material for accident mitigation and are not considered to be a source of any postulated accident. The analysis verifies that a fuel handling accident (FHA) occurring at least 100 hours after being critical in a reactor core will not result in dose consequences above the regulatory limits without the containment closure function provided by the CBED [containment building equipment door]. The applicability and action for the CBED will not be changed when movement of recently irradiated fuel is in progress and this function ensures acceptable dose consequences. Therefore, the proposed change will not increase the probability of an accident because the penetration function has not been altered and this function is not a potential source for accidents. Additionally, the proposed change will not significantly increase the consequences of an accident because the analysis has verified that dose consequences will be maintained less than the required regulatory limits.

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

The proposed change only modifies when containment building penetrations need to be available for accident mitigation and does not alter their function, design, or operation. These penetrations only serve to minimize the release of radioactive material in the event of postulated accidents and do not have the potential to create an accident. Since the function of the penetrations is not being changed and they do not have an accident generation potential, the possibility of a new or different kind of accident is not created.

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

The proposed change will not alter the function, design, or operation of the containment building penetrations for postulated accidents that require this feature for the mitigation of the event. The analysis has determined that the CBED availability can be limited to those activities that involve the movement of irradiated fuel that has been in a critical reactor core within the previous 100 hours. Therefore, not requiring the CBED to be available 100 hours or longer afterwards will not impact plant safety or result in dose consequences above established regulatory limits. The proposed change will not alter any setpoints or other functions that serve to maintain the safety limits. Therefore, the proposed change 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of amendment request: January 14, 2003.

Description of amendment request: The proposed amendment will revise the Yankee Rowe Nuclear Power Station License and Technical Specifications to delete operational and administrative requirements that would no longer be required once the spent nuclear fuel has been transferred from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI).

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

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

No. The proposed changes reflect the complete transfer of all spent nuclear fuel from the Spent Fuel Pit to the Independent Spent Fuel Storage Installation (ISFSI). Design basis accidents related to the Spent Fuel Pit are discussed in the YNPS FSAR. These postulated accidents are predicated on spent nuclear fuel being stored in the Spent Fuel Pit. With the removal of the spent fuel from the Spent Fuel Pit, there are no remaining important to safety systems required to be monitored and there are no remaining credible accidents that require that actions of a Certified Fuel Handler or non-Certified Fuel Handler to prevent occurrence or mitigate the consequences.

The YNPS FSAR provides a discussion of radiological events postulated to occur as a result of decommissioning with the bounding consequence resulting from a materials handling event. The proposed changes do not have an adverse impact on decommissioning activities or any of their postulated consequences.

The proposed change to the Design Features section of the Technical Specifications clarifies that the spent fuel is being stored in dry casks within an ISFSI. The probability or consequences of accidents at the ISFSI are evaluated in the dry cask vendor's FSAR and are independent of the accidents evaluated in the YNPS FSAR.

Based on the above, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed changes reflect the reduced operational risks as a result of the spent nuclear fuel being transferred to dry casks within an ISFSI. The proposed changes do not modify any physical systems, or components. The plant conditions for which the YNPS FSAR design basis accidents relating to spent fuel have been evaluated are no longer applicable. The aforementioned proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of an accident. Design basis accidents associated with the dry cask storage of spent fuel are already considered in the dry cask system's Final Safety Analysis Report. No new accident scenarios are created as a result of deleting non-applicable operational and administrative requirements. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. As described above, the proposed changes reflect the reduced operational risks as a result of the spent nuclear fuel being transferred to dry casks within an ISFSI. The design basis and accident assumptions within the YNPS FSAR and the Defueled Technical Specifications relating to spent fuel are no longer applicable. The proposed changes do not affect remaining plant operations, systems, or components supporting decommissioning activities. In addition, the proposed changes do not result in a change in initial conditions, system response time, or in any other parameter affecting the course of a decommissioning activity accident analysis. Therefore, the proposed changes will not involve a significant reduction in the margin of safety.

Based on the considerations noted above, it is concluded that the proposed changes will not endanger the public health and safety.

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

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Section Chief: Scott W. Moore.

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.

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

Date of amendment request: January 16, 2003.

Brief description of amendment request: The proposed amendment would revise the applicable Technical Specifications requirements for rod position monitoring during the current operating cycle (Cycle 22) to allow the use of an alternate method of determining rod position. This would be effective until repair of the indication system can be completed during the next shutdown of sufficient duration.

*Date of publication of individual notice in **Federal Register:*** January 24, 2003 (68 FR 3566).

Expiration date of individual notice: February 7, 2003, for comments; February 24, 2003, for hearings.

Florida Power and Light Company, Docket No. 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: November 26, 2002.

Brief description of amendments: The proposed license amendments would revise Technical Specifications (TSs) to increase the total spent fuel wet storage capacity by adding a spent fuel storage rack in the cask area in each unit's spent fuel pool. Also, it would revise the location called out in the Design Features sections 5.6.1.1a and b of the TSs referring to Updated Final Safety Analysis Report Appendix 14D, rather than referring to Westinghouse Report WCAP–14416–P.

*Date of publication of individual notice in the **Federal Register:*** January 28, 2003 (68 FR 4246).

Expiration date of individual notice: February 27, 2003.

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.*

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: May 29, 2001, and its supplements dated August 29, 2001, and September 24, 2002.

Brief description of amendment: The amendment revises paragraph 2.C.(5), ''Physical Protection,'' of Facility Operating License No. DPR–61 to reference the Defueled Physical Security Plan that includes the security plan for the Independent Spent Fuel Storage Installation.

Date of issuance: January 30, 2003.

Effective date: January 30, 2003, and shall be implemented within 30 days from the date of issuance and prior to the transfer of spent nuclear fuel to the Independent Spent Fuel Storage Installation.

Amendment No.: 199.

Facility Operating License No. DPR–61: The amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44163). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 30, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 8, 2002, as supplemented October 23, 2002.

Brief description of amendment: The amendment authorized changes to the Updated Final Safety Analysis Report (USFAR) for Fermi 2 by allowing implementation of the Boiling Water Reactor Vessel and Internals Project reactor pressure vessel Integrated Surveillance Program as the basis for demonstrating the compliance with the requirements of Appendix H, ''Reactor Vessel Material Surveillance Program Requirements,'' to 10 CFR part 50.

Date of issuance: January 30, 2003.

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

Amendment No.: 152.

Facility Operating License No. NPF–43: Amendment authorizes changes to the USFAR.

Date of initial notice in ***Federal Register:*** September 3, 2002 (67 FR 56320). The October 23, 2002, supplemental letter provided additional clarifying information that did not change the original no significant hazards consideration determination or expand the amendment beyond the scope of the original notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 30, 2003.

No significant hazards consideration comments received: No.

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: April 18, 2002, as supplemented August 7, and October 9 and October 30, 2002, and January 15, 2003.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.7.15 in response to Boraflex degradation to provide revised spent fuel pool (SFP) storage criteria, and revised fuel enrichment and burnup requirements that take credit for soluble boron. TS 4.3.1 is revised to increase the required soluble boron credit from a concentration of 730 parts per million (ppm) to 850 ppm to ensure acceptable levels of subcriticality in the SFPs. Associated changes to the TS Bases are also included.

Date of issuance: February 4, 2003.

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

Amendment Nos.: 210 & 191.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42820). The supplements dated August 7, and October 9 and October 30, 2002, and January 15, 2003, provided clarifying information that did not change the scope of the April 18, 2002, 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 February 4, 2003.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: October 22, 2002.

Brief description of amendment: The amendment deletes TS 5.5.3, ''Post Accident Sampling System (PASS),'' and thereby eliminates the requirements to have and maintain the PASS at Columbia Generating Station. The amendment also addresses related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment,'' and License Condition 2.C.(13), ''Post Accident Sampling.''

Date of issuance: January 27, 2003.

Effective date: January 27, 2003, to be implemented within 60 days from the date of issuance.

Amendment No.: 184.

Facility Operating License No. NPF–21: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** December 24, 2002 (67 FR 78518). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 2003.

No significant hazards consideration comments received: No.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 14, 2002, as supplemented by letters dated July 9, August 2, September 16, and November 7 and 22, 2002.

Brief description of amendment: This amendment increases the licensed power level by approximately 1.7 percent from 3,039 megawatts thermal (MWt) to 3,091 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

Date of issuance: January 31, 2003.

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

Amendment No.: 129.

Facility Operating License No. NPF–47: The amendment revised the Facility Operating License and Technical Specifications.

Date of initial notice in ***Federal Register:*** June 11, 2002 (67 FR 40022). The July 9, August 2, September 16, and November 7 and 22, 2002, supplemental letters 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 31, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 24, 2001, as supplemented on May 22, 2002.

Brief description of amendment: The amendment revised information in the

Final Safety Analysis Report regarding the protection of the component cooling water (CCW) system from natural phenomena. The change addresses the fact that a portion of one safety-related loop of the CCW system is routed through the fuel storage building, where the structure was not designed to protect the CCW piping from the effects of natural phenomena.

Date of issuance: January 27, 2003.

Effective date: January 27, 2003.

Amendment No.: 214.

Facility Operating License No. DPR–64: Amendment revised the Final Safety Analysis Report.

*Date of initial notice in **Federal Register:*** October 3, 2001 (66 FR 50466). The May 22, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 27, 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: February 26, 2002, as revised by letters dated October 9 and 30, 2002.

Brief description of amendment: The amendment revises the definition of Operable in Technical Specification (TS) 1.0.K with respect to support system requirements for alternating current power sources. Conforming changes are also made to a specific support system TS in Sections 3/4.5, ''Core and Containment Cooling Systems'', 3/4.7, ''Station Containment Systems'', and 3/4.10, ''Auxiliary Electrical Power Systems,'' and associated Bases.

Date of Issuance: February 4, 2003.

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

Amendment No.: 213.

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

*Date of initial notice in **Federal Register:*** (67 FR 78519). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated February 4, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 19, 2002, as supplemented December 26, 2002.

Brief description of amendments: The amendments add a new analytical method to Technical Specifications (TS) section 5.6.5, ''Core Operating Limits Report.'' The change supports the core design efforts used for the Unit 2 refueling outage which began on January 21, 2003.

Date of issuance: February 4, 2003.

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

Amendment Nos.: 159 & 145.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 15, 2002 (67 FR 63694). The December 26, 2002, supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 4, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 31, 2002, as supplemented by letter dated October 16, 2002.

Brief description of amendments: These amendments revised Technical Specifications (TSs) 3.8.2.1, ''DC Sources—Operating,'' and 3.8.2.2, ''DC Sources—Shutdown''; and added the new Specification 6.8.4.i, ''Battery Monitoring and Maintenance Program.'' The changes also included the relocation of the following TS items to a licensee-controlled program: (1) A number of surveillance requirements that require the performance of preventive maintenance, and (2) certain battery and battery cell parameter values that are periodically verified to monitor early indications of DC subsystem degradation.

Date of issuance: January 29, 2003.

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

Amendment Nos.: 164 and 126.

Facility Operating License Nos. NPF–39 and NPF–85. The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 17, 2002 (67 FR 58643). The supplement dated October 16, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois

Date of application for amendments: February 28, 2001, as supplemented by letter dated June 13, 2002.

Brief description of amendments: Revise the Technical Specifications to eliminate the requirement for at least one person qualified to stand watch to be present in the control room when nuclear fuel is stored in the spent fuel pool.

Date of issuance: January 31, 2003.

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

Amendment Nos.: 183 and 170.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 27, 2001 (66 FR 34283).

The June 13, 2002, 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 amendments is contained in a Safety Evaluation dated January 31, 2003.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Beaver County, Pennsylvania

Date of application for amendment: May 31, 2002, as supplemented July 19, and September 3, 2002.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.1.1.4, upper limit for the moderator temperature coefficient (MTC), from 0×10^{-4} change in reactivity per degree Fahrenheit ($\Delta k/k/°F$) to $+0.2 \times 10^{-4}$ $\Delta k/k/°F$ for power levels up to 70 percent of rated thermal power (RTP), and ramping linearly to 0 $\times 10^{-4}$ Δk/k/°F from 70 percent to 100 percent RTP. The change is needed to address future core designs with higher energy requirements, associated with plant operation at higher capacity factors.

Date of issuance: February 6, 2003.

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

Amendment No.: 251.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 17, 2002 (67 FR 58644). The July 19, and September 3, 2002, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 2003.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of application for amendment: March 27, 2002, as supplemented on October 7, 2002.

Brief description of amendment: The amendment revises the Technical Specifications section 3.6.3, ''Emergency Power Sources,'' to extend the current allowable outage time for an inoperable diesel generator from 7 days to 14 days, and section 3.4.4, ''Emergency Ventilation System,'' and section 3.4.5, ''Control Room Air Treatment System,'' to reflect the change to section 3.6.3.

Date of issuance: February 3, 2003.

Effective date: February 3, 2003.

Amendment No.: 179.

Facility Operating License No. DPR–63: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 30, 2002 (67 FR 21290). The October 7, 2002, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 25, 2002. The application was initially submitted to the Nuclear Regulatory Commission with an incorrect date of April 25, 2001. The Nuclear Management Company, LLC, subsequently submitted a letter dated May 30, 2002, correcting the date of the application as April 25, 2002.

Brief description of amendment: The amendment changes Technical Specification (TS) 3.7/4.7, ''Containment Systems,'' to allow the use of 10 CFR part 50, Appendix J, Option B, for Types B and C containment leak rate testing and adds a new TS section 6.8.M, ''Programs and Manuals—Primary Containment Leakage Rate Testing Program.''

Date of issuance: February 4, 2003.

Effective date: As of the date of issuance and to be implemented within 75 days.

Amendment No.: 132.

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

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56325).

The May 30, 2002, letter corrected the date of the application and did not change the NRC staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 4, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 25, 2002, as supplemented by letter dated October 23, 2002.

Brief description of amendments: These amendments revise the Susquehanna Steam Electric Station Final Safety Analysis Report (SSES FSAR) by replacing the current plant-specific reactor pressure vessel material surveillance program with the Boiling Water Reactor Integrated Surveillance Program.

Date of issuance: February 6, 2003.

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

Amendment Nos.: 208 and 182.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the SSES FSAR.

*Date of initial notice in **Federal Register:*** September 3, 2002 (67 FR 56328). The October 23, 2002, supplemental letter provided additional information that clarified the application, but did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 6, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 23, 2002.

Brief description of amendment: The amendment updates the reference to 10 CFR 20.203 with the corresponding reference to 10 CFR 20.1601. Hope Creek Generating Station Technical Specification (TS) 6.12, ''High Radiation Area,'' is revised to be consistent with the Standard TSs, General Electric Plants (NUREG–1433, Rev. 2).

Date of issuance: January 30, 2003.

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

Amendment No.: 142.

Facility Operating License No. NPF–57: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 10, 2002 (67 FR 75884).

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

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: August 21, 2002.

Brief description of amendments: The amendments revise the technical specifications (TSs) to replace reference to specific valves for preventing uncontrolled boron dilution. The revised TSs incorporate a general statement for preventing uncontrolled boron dilution, consistent with the improved standard TSs.

Date of issuance: January 27, 2003.

Effective date: January 27, 2003.

Amendment Nos.: Unit 1–149; Unit 2–137.

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

Date of initial notice in ***Federal Register:*** October 1, 2002 (67 FR 61686).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 27, 2003.

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 23, 2002.

Brief description of amendments: The amendments relocated the shutdown margin limits to the Core Operating Limits Report and modified certain boration requirements consistent with NUREG–1431. The amendments also correct some typographical errors in the Technical Specification pages.

Date of issuance: February 4, 2003.

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

Amendment Nos.: Unit 1–150; Unit 2–138.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42830).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 4, 2003.

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: November 6, 2002.

Brief description of amendments: The amendments revised the Browns Ferry Nuclear Plant, Units 2 and 3, Updated Final Safety Analysis Report (UFSAR) to modify the basis for TVA's compliance with the requirements of Appendix H to title 10 of the Code of Federal Regulations part 50, ''Reactor Vessel Material Surveillance Program Requirements.''

Date of issuance: January 28, 2003.

Effective date: As of the date of issuance, to be incorporated into the UFSAR at the time of its next update.

Amendment Nos.: 279 & 238.

Facility Operating License Nos. DPR–52 and DPR–68: Amendments revised the UFSAR.

Date of initial notice in ***Federal Register:*** November 26, 2002 (67 FR 70770). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 28, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 3, 2002, as supplemented October 17, 2002, and January 29, 2003.

Brief description of amendments: The amendments revise Technical Specification (TS) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. Thise changes to SR 4.0.3 will allow an extension of up to 24 hours or the limit of the surveillance frequency, whichever is greater. The amendments also include editorial changes to make the revised TS consistent with the Standard TS for Westinghouse plants. In addition, the amendments include the adoption of the TS Bases Control Program listed in NUREG–1431, Revision 2.

Date of issuance: February 5, 2003.

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

Amendment Nos.: 280 and 271.

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

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68745). The January 29, 2003, supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2003.

No significant hazards consideration comments received: No.

Dated in Rockville, Maryland, this 10th day of February, 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–3689 Filed 2–13–03; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Summission for OMB Review: Comment Request Review of Expiring Information Collection: OPM 1647

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Pub. L. 104–13, May 22, 1995), this notice announces that the Office of Personnel Management submitted a request for renewal of authorization for an information collection to the Office of Management and Budget. OPM Form 1647, Combined Federal Campaign Eligibility Application, is used to review the eligibility of national, international, and local charitable organizations that wish to participate in the Combined Federal Campaign.

We estimate 1,400 OPM Forms 1647 will be completed annually. Each form takes approximately three hours to complete. The annual estimated burden is 4,200 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–2150, Fax (202) 418–3251 or E-mail to *mbtoomey@opm.gov.* Please include a mailing address with your request.

DATES: Comments on this proposal should be received by March 20, 2003.

ADDRESSES: Send or deliver comments to:

Curtis Rumbaugh, Office of CFC Operations, U.S. Office of Personnel Management, 1900 E Street, NW., Room 5450, Washington, DC 20415; and

Stuart Shapiro, OPM Desk Officer, Office of Information & Regulatory Affairs, Office of Management and Budget, New Executive Office Building, NW., Room 10235, Washington, DC 20503.

Office of Personnel Management.

Kay Coles James,

Director.

[FR Doc. 03–3819 Filed 2–14–03; 8:45 am]

BILLING CODE 6325–46–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. IC–25931; File No. 812–12881]

Vision Group of Funds, *et al.*; Notice of Application

February 10, 2003.

AGENCY: The Securities and Exchange Commission (''SEC'' or the ''Commission'').