alternative(s) the LSNARP will recommend or endorse to the Commission.

FOR FURTHER INFORMATION CONTACT: U.S. Nuclear Regulatory Commission, Atomic Safety and Licensing Board Panel, Mail Stop T–3 F23, Washington, DC 20555–0001; Attn: John C. Hoyle (telephone 301–415–7467; e-mail JXH5@NRC.GOV) or Jack G. Whetstine (telephone 301–415–7391; e-mail JGW@NRC.GOV).

Public Participation: Interested persons may make oral presentations to the LSNARP or file written statements. An oral presentations request should be made to one of the contact persons listed above as far in advance as practicable so that appropriate arrangements can be made.

Dated: December 22, 1999.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 99–33778 Filed 12–28–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting; Notice

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of December 27, 1999, January 3, 10, and 17, 2000.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of December 27

There are no meetings scheduled for the Week of December 27.

Week of January 3—Tentative

Wednesday, January 5

9:55 a.m.
Affirmation Session (Public Meeting) (if needed)

Week of January 10—Tentative

Monday, January 10

10:00 a.m.
Meeting with D.C. Cook (Public Meeting) (Contact: John Stang, 301–415–1345)

Tuesday, January 11

9:30 a.m.
Briefing on Status of Research Programs, Performance, and Plans (including Status of Thermo-Hydraulics) (Public Meeting) (Contact: Jocelyn Mitchell, 301–415–5289)

Wednesday, January 12

9:55 a.m.
Affirmation Session (Public Meeting) (if needed)

10:00 a.m.
Briefing on Status of NRR Programs, Performance, and Plans (Public Meeting) (Contact: Mike Case, 301–415–1134)

Week of January 17—Tentative

Wednesday, January 19

9:30 a.m.
Discussion of Management Issues (Closed—Ex. 2 & 6)

Thursday, January 20

9:55 a.m.
Affirmation Session (Public Meeting) (if needed)

10:00 a.m.
Briefing on Status of CIO Programs, Performance, and Plans (Public Meeting) (Contact: Donnie Grimsley, 301–415–8702)

Friday, January 21

10 a.m.
Briefing on Native American, State of Nevada, and Affected Units of Local Governments Representratives Responses to DOE's Draft Environmental Impact Statement (EIS) for a proposed HLW Geologic Repository (Public Meeting)

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings can (Recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMAITON: Bill Hill (301) 415–1661.

ADDITIONAL INFORMATION: By a vote of 5–0 on December 22, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's Rules that ''Affirmation of GPU Nuclear Corporation, Docket No. 50–219, OLA–2, Memorandum and Order Terminating Proceeding), LBP 99–45 (Dec 15, 1999)'' and ''Affirmation of Niagara Mohawk Power Corp. et al. (Nine Mile Point, Units 1 & 2), Docket Nos. 50–220 and 50–410'' (PUBLIC MEETING) be held on December 22, and on less than one week's notice to the public.

* * * * *

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

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). 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 wmh@nrc.gov or dkw@nrc.gov.

Dated: December 23, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–33890 Filed 12–23–99; 4:34 pm]

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 December 4, 1999, through December 17, 1999. The last biweekly notice was published on December 15, 1999 (64 FR 70077).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 28, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: November 22, 1999.

Description of amendments request: The proposed amendment revises Technical Specification (TS) 5.5.11, ''Ventilation Filter Testing Program'' for laboratory testing of charcoal in Clavert Cliffs engineered safety feature (ESF) ventilation systems to reference the latest charcoal testing standard (American Society for Testing and Materials [ASTM] D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon''). This TS change was requested by the Nuclear Regulatory Commission (NRC) in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' and is based on the NRC's determination that testing nuclear-grade activated charcoal to standards other than ASTM D3803–1989 does not provide assurance for complying with the current licensing basis as it relates to the dose limits of General Design Criterion 19 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR) and Subpart A of 10 CFR Part 100. The generic letter provided a sample TS that the NRC considers acceptable. The proposed revision to TS 5.5.11 meets the intent of the sample TS. Specifically, the proposed change removes the reference to testing in accordance with American National Standards Institute N510–1975 and changes the allowable methyl iodide penetration to an acceptance criterion that is derived from applying a safety factor of two to the charcoal filter efficiency assumed in Calvert Cliffs design basis dose analysis. The proposed changes will ensure that the charcoal filters used in ESF ventilation systems will perform in a manner that is consistent with the particular ESF charcoal adsorption efficiencies assumed in the analyses of design basis accidents.

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.

This proposed change makes changes to the methods, test conditions, and acceptance criteria associated with the performance of the laboratory tests of charcoal samples. The effected equipment is used to mitigate the consequences of an accident and are not accident initiators. This proposed change does not make any changes to the method of obtaining the charcoal sample. No structural changes or modifications are being made to the ESF ventilation equipment. This proposed change does not make any changes to equipment, procedures, or processes that increase the likelihood of an accident. Therefore, this proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The ESF ventilation systems are designed to mitigate the consequences of accidents. The design basis analysis of the accidents account to varying degrees for the reduction in airborne radioactive material provided by the charcoal filters. The proposed change will change the charcoal filter test protocol to ASTM D3803–1989. The use of this standard will produce more accurate and reproducible laboratory test results and provides a more conservative estimate of charcoal filter capability. The proposed change makes changes to the methyl iodide penetration acceptance criteria to ensure that the charcoal filters are capable of performing their required safety function for the expected operating cycle. The proposed change will make it more likely that the charcoal will meet its intended safety function as described in the Updated Final Safety Analysis Report. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

Based on the above, the proposed change does 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 type of accident from any accident previously evaluated.

The proposed change will not make any physical changes to the plant or changes to the ESF ventilation system operation. The proposed change is limited to the ESF ventilation system testing protocol, test conditions, and acceptance criteria. These changes are administrative in nature. This proposed change does not make any changes to the method of obtaining the charcoal sample. This proposed change does not cause any ESF ventilation equipment to be operated in a new or different manner. No structural changes or modifications are being made to the ESF ventilation equipment. This proposed change does not create any new interactions between any plant components. Therefore, the possibility of a new or different type of accident is not created by this proposed change.

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

The safety function of the ESF ventilation systems is to mitigate the consequences of accidents by reducing the potential release of radioactive material to the environment or the Control Room following a design basis accident. The TS requirements for laboratory testing of charcoal samples provides assurance that the charcoal filters in these systems are capable of reducing airborne radioactive material to within acceptable limits. The proposed license amendment requires the use of the latest NRC-accepted charcoal testing standard and makes changes to the charcoal testing methyl iodide removal efficiency acceptance limits in accordance with the formula provided by the NRC in Generic Letter 99–02. The proposed license amendment continues to provide assurance that the charcoal filters are capable of reducing airborne radioactive material to within acceptable limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Sheri R. Peterson.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: November 22, 1999.

Description of amendments request: The Baltimore Gas and Electric Company (BGE) requests an amendment to implement a change to the Calvert Cliffs Nuclear Power Plant (CCNPP) Updated Final Safety Analysis Report (UFSAR) that constitutes an unreviewed safety question as described in 10 CFR 50.59.

The change revises the information currently provided within the UFSAR on aircraft and their flight paths for Patuxent River Naval Air Station (Pax River NAS). The existing information is outdated and does not reflect current conditions for aircraft utilizing Pax River NAS. Additionally, the UFSAR will be revised to add information pertaining to the corporate helipad located northwest of the plant.

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 probability of an aircraft crash was not quantified during the timeframe of licensing and construction of the plant. As was noted previously, the Directorate of Licensing at the U.S. Atomic Energy Commission concurred with Baltimore Gas and Electric Company's conclusion that no special design provisions were required to be incorporated into Calvert Cliffs Nuclear Power Plant (CCNPP) because the probability of an aircraft crash affecting the plant was acceptably low (implies a probability of less than 10^{-7}/Year). Therefore, the probability of an aircraft crash affecting the plant was acceptably low at less than 10^{-7}/year.

The probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines was considered to still be below the Standard Review Plan (SRP) (NUREG–0800) level of acceptability of 1.0×10^{-7} per year for CCNPP. The probability of an aircraft accident during the timeframe of original construction and licensing of the plant was never quantified. Since today's probability of an aircraft accident may be higher based on the fact that, at times, aircraft going into Patuxent River Naval Air Station fly over the plant, where previously they came no closer than seven miles from the plant (as described in the UFSAR), the probability of occurrence of an accident will conservatively be considered to have increased. However, it should be noted that the probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is still considered to be below 1.0×10^{-7} cr/yr, which is acceptable since it is within SRP Section 3.5.1.6 guidelines. Since the above probability of an aircraft accident meets the criteria of SRP Section 3.5.1.6, no additional design or procedural protection is required. Note that the SRP criteria is only being used as one acceptable method of evaluating risk. Use of this method is not a commitment to the SRP and does not incorporate the SRP into our licensing basis.

Changes to the aircraft flight patterns and/or frequency (probability) have no affect on the design or method of operating equipment necessary to mitigate the consequences of previously analyzed accidents. As was noted above, the aircraft hazard was considered to be acceptable and, therefore, no additional design or procedural protection is required for the plant. Since the aircraft hazard is considered acceptable (where additional design features are not required), it can be concluded that no action assumed to occur within the accident analysis of CCNPP's Updated Final Safety Analysis Report Chapter 14 will be degraded or prevented. Therefore, it is concluded that the current calculated aircraft hazard will not result in an increase of the consequences of an accident preciously evaluated in the UFSAR.

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

All possible malfunctions have been previously analyzed. Aircraft hazard was addressed within the original design of the plant. The frequency/probability of an aircraft crash was considered to be so low that special design provisions to protect against aircraft crashes did not have to be considered during construction of CCNPP. The current calculated aircraft hazard is considered to still be within SRP Section 3.5.1.6 guidelines. The possibility for a malfunction of a different type than preciously evaluated in the UFSAR is not created.

Aircraft accidents were considered within the original plant design. The probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines is still considered to be below the level of acceptability (per SRP Section 3.5.1.6) and no special design provisions are required. Since an aircraft crash is not a design basis concern, it is not plausible that the possibility of a new accident is created that has not been previously evaluated in the UFSAR. There are also no new challenges to safety-related equipment.

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

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

The probability of an aircraft crash affecting the plant, at the time of original licensing and construction, was so low that no special design provisions were needed in the plant for such an event. Since aircraft hazards did not have to be considered within the design of the plant, no margin of safety was required or established for such a hazard. All of the plant equipment and initial condition assumptions stipulated within the UFSAR Chapter 14 accident analysis would not be affected by such an event.

The calculated probability of an aircraft accident resulting in radiological consequences greater than 10 CFR Part 100 exposure guidelines, based on today's aircraft hazard, is considered to be below the 1.0×10^{-7} per year stipulated within SRP Section 3.5.1.6. Therefore, there is still no need for special design provisions within the plant to guard against such an event. All of the plant equipment and initial condition assumptions stipulated within the UFSAR Chapter 14 accident analysis remain unchanged. The plant will continue to operate in such a manner that will ensure acceptable levels of protection for the health and safety of the public.

Therefore, this proposed change does not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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: Sheri R. Peterson.

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

Date of amendments request: November 23, 1999

Description of amendments request: The requested amendments would change Technical Specification (TS) 5.5.7.c.1, ''Ventilation Filter Testing.'' The testing criteria would be changed consistent with the NRC request in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

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

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

The proposed amendment revises TS 5.5.7.c.1 to require testing of the SGT [Standby Gas Treatment] system charcoal in accordance with American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.'' Per the existing TSs, the SGT system charcoal must meet an acceptance criteria of < 1.0% penetration of methyl iodide when tested at a relative humidity ≥ 70%. CP&L performs this testing in accordance with the criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, 1976, ''Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.'' As stated in Updated Final Safety Analysis Report, Section 6.5.1.1, the purpose of the SGT system, along with that of the primary and secondary containment, is to mitigate accident consequences. It is not associated with any initiating events and, therefore, cannot affect the probability of any accident.

ASTM D3803–1989 is an industry accepted standard for charcoal filter testing. The conditions employed by this standard were selected to approximate operating or accident conditions of a nuclear reactor which would severely reduce the performance of activated carbons. The key difference associated with the two testing protocols is the testing temperature. Specifically, testing to a challenge temperature of 30 °C per ASTM D3803–1989 versus 80 °C per Regulatory

Guide 1.52 results in a much more stringent test. Testing at a higher temperature tends to eliminate impurities and moisture from the sample. This creates the possibility of the charcoal achieving a slightly higher efficiency than actual. Other parameter changes will not significantly affect charcoal test performance and will result in more accurate and reproducible test results.

The proposed TS change also includes a requirement that the test be performed with a face velocity of 61 fpm. A single BSEP SGT system train operates at a maximum flow rate of 4200 scfm which corresponds to a face velocity of 61 fpm. In accordance with Generic Letter (GL) 99–02, this requirement has been included in TS 5.5.7.c.1.

As recommended by GL 99–02, the proposed amendment incorporates a safety factor of 2 into the allowed methyl iodide penetration limit. The existing TS 5.5.7.c.1 acceptance criteria of 99% does not account for a safety factor. In previous testing, CP&L has applied the safety factor provided by Regulatory Guide 1.52, Revision 1, 1976, to the laboratory testing results to ensure proper charcoal performance. The proposed changes to TS 5.5.7.c.1 require that charcoal samples, tested in accordance with the methodology of ASTM D3803–1989, show the methyl iodide penetration to be < 0.5%. The 0.5% penetration limit is derived by applying a safety factor of 2 to the 99% filtration efficiency assumed in the current bounding calculations for offsite radiological dose release limits. As such, the acceptance criteria of < 0.5% penetration of methyl iodide ensures that 10 CFR 100 offsite dose limits are not exceeded.

Based on the more stringent testing temperature requirements, the new face velocity testing requirement, and the acceptance criteria of < 0.5% penetration of methyl iodide, the proposed change will not result in an increase in the consequences of an accident previously evaluated.

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

The proposed changes revise the required testing methodology for SGT system charcoal. The SGT system is not an initiator of any accident, and no new accident precursors are created due to the change in the charcoal testing methodology. In addition, the change does not alter the design, function, or operation of the SGT system. Therefore, the proposed change to test SGT system charcoal in accordance with ASTM D3803–1989 will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment upgrades the SGT system charcoal testing requirements to those contained in ASTM D3803–1989. The conditions employed by ASTM D3803–1989 were selected to approximate operating or accident conditions of a nuclear reactor which could reduce the performance of activated carbons. The key difference between CP&L's current testing protocol and ASTM D3803–1989 is the testing temperature. Specifically, testing to a challenge temperature of 30°C per ASTM D3803–1989 versus 80°C per Regulatory Guide 1.52 results in a much more stringent test.

The proposed TS change also includes a requirement that the test be performed with a face velocity of 61 fpm. A single BSEP SGT system train operates at a maximum flow rate of 4200 scfm which corresponds to a face velocity of 61 fpm. In accordance with GL 99–02, this requirement has been included in TS 5.5.7.c.1.

As recommended by GL 99–02, the proposed amendment incorporates a safety factor of 2 into the allowed methyl iodide penetration limit. The existing TS 5.5.7.c.1 acceptance criteria of 99% does not account for a safety factor. In previous testing, CP&L has applied the safety factor provided by Regulatory Guide 1.52, Revision 1, 1976, to the laboratory testing results to ensure proper charcoal performance. The proposed changes to TS 5.5.7.c.1 require that charcoal samples, tested in accordance with the methodology of ASTM D3803–1989, show the methyl iodide penetration to be < 0.5%. The 0.5% penetration limit is derived by applying a safety factor of 2 to the 99% filtration efficiency assumed in the current bounding calculations for offsite radiological dose release limits. As such, the acceptance criteria of < 0.5% penetration of methyl iodide ensures that 10 CFR 100 offsite dose limits are not exceeded.

Based on the more stringent testing temperature requirements, the new face velocity testing requirement, and the acceptance criteria of < 0.5% penetration of methyl iodide, 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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: November 30, 1999.

Description of amendment request: The amendment revises Technical Specifications (TS) Section 5.5.11, Ventilation Filter Testing Program (VFTP) testing requirements. The proposed change requires VFTP testing be done according to ASTM D3803–1989 protocol in lieu of previous standards.

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:

Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The CP&L conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

The proposed change to Technical Specification Section 5.5.11, "Ventilation Filter Testing Program," does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change updates the required testing of Engineered Safety Features (ESF) ventilation filter systems to more recent standards accepted by the NRC and described in Generic Letter (GL) 99–02, "Laboratory Testing of Nuclear-Grade Activated Charcoal." The NRC has found that charcoal filter test protocols other than American Society for Testing and Materials (ASTM) standard ASTM D3803–1989 do not assure accurate and reproducible test results. Since this proposed change references an acceptable testing standard and provides assurance that the current licensing basis is met, the proposed change does not involve an increase in the probability 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 evaluated?

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. The proposed change introduces a new testing standard for ESF ventilation system charcoal samples removed for testing and does not involve manipulation of plant systems to perform the charcoal test. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change revises the required testing standard for ESF ventilation charcoal filter systems and does not alter plant design margins or analysis assumptions as described in the Updated Final Safety Analysis Report. The proposed change does not affect any limiting safety system setpoint, calibration method, or setpoint calculation. The proposed change is more restrictive with regard to testing protocol and less restrictive with respect to the allowed penetration during testing of the Control Room ventilation system charcoal. However, the allowed increase in penetration is in accordance with the method for determining the allowable penetration described in GL 99–02. 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Section Chief: Richard P. Correia.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: November 18, 1999.

Description of amendment request: The proposed amendment requests a revision to Technical Specification (TS) 5.5.7.c. The changes would revise the requirements that (1) a sample of the charcoal absorber for the standby gas treatment (SGT) system and the control room emergency filtration (CREF) system be tested in accordance with American Society for Testing and Materials (ASTM) D3803–1986, ''Standard Test Method for Nuclear-Grade Activated Carbon'', (2) methyl iodide penetration be less than a value of .175% for the SGT system and 1.0% for the CREF system, and (3) charcoal absorber testing be conducted at a relative humidity of greater than or equal to 70%. As requested by Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' Energy Northwest proposed that TS 5.5.7.c be revised so that (1) testing of charcoal absorber samples be in accordance with ASTM D3803–1989 at a specified temperature of 30° Centigrade (C) [86° Fahrenheit (F)], (2) methyl iodide penetration to be less than a value of 0.5% for the SGT system and 2.5% for the CREF system, (3) testing be performed at 70% relative humidity, and (4) a face velocity of 75 feet-per-minute (fpm) will be specified for the SGT system. In addition, the revision to TS 5.5.7.c will note that variations in testing parameters are permitted in accordance with the guidance in Table 1 and Section A5.2 of ASTM D3803–1989.

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

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

The SGT System is designed to limit the release of airborne radioactive contaminants from secondary containment to the atmosphere within the guidelines of 10 CFR 100 in the event of a DBA [design basis accident]. The CREF System provides a radiologically controlled environment from which the plant can be safely operated following a DBA. The proposed amendment will require that charcoal from these two ESF [engineered safeguard feature] systems be tested to the more conservative standards of ASTM D3803–1989. Using the more conservative ASTM D3803–1989 testing standard will provide no increase in the probability of an accident previously evaluated.

The staff considers ASTM D3803–1989 to be the most accurate and most realistic protocol for testing charcoal in ESF ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Using the more conservative ASTM D3803–1989 testing standard will provide greater assurance that the ESF ventilation systems will properly perform their safety function, thus assuring no increase in the radiological consequences of a DBA.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change will not create a new or different kind of accident since it only requires that charcoal from the SGT and CREF safety-related filtration systems be tested to the more conservative standards of ASTM D3803–1989. Using the more conservative ASTM D3803–1989 testing standard will provide even greater assurance that the ESF ventilation systems will properly perform their safety function, thus helping to minimize the radiological consequences of a DBA. The increased margin provided by the more conservative testing standard will assure no new or different kinds of accidents results from the proposed change.

Therefore, the operation of WNP–2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment requires that more conservative ESF charcoal filter testing criteria be used to verify ESF ventilation systems are operable. More conservative testing criteria will provide greater assurance that the ESF ventilation systems will properly perform their safety function, thus helping to minimize the radiological consequences of a DBA. Using more conservative testing criteria will result in maintaining the current 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, N.W., Washington, D.C. 20005–3502.

NRC Section Chief: Stephen Dembek.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County,Washington

Date of amendment request: November 18, 1999.

Description of amendment request: The proposed amendment requests a revision to subsection 4.3.1.2.b of Technical Specification 4.3, Fuel Storage. The change would revise the current wording, which describes the spacing of the fuel in the new fuel racks, with wording that would limit the number of fuel assemblies that may be stored in the facility and establish increased spacing limitations for storage of new fuel assemblies in the racks.

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

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

The proposed change does not increase the consequences of any previously analyzed accident or transient, since the arrangement of new nuclear fuel in storage racks maintains the effective neutron multiplication factor much less than 0.95. The change in configuration requirements will not increase the probability of any previously analyzed accident, because physical constraints are installed in the storage racks when new fuel assemblies are inserted, assuring that only certain cells can be used for storage of new fuel.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is consistent with a new fuel criticality analysis performed in support of a previously implemented change to Section 9.1 of the FSAR. A variety of accidents were considered in that analysis, and it was determined that the effective neutron multiplication factor was well below specified limits for any normal or accident case.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident previously evaluated.

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

The current wording of Technical Specification 4.3.1.2.b was determined to not provide sufficient margin of safety to assure that the requirements of Technical Specification 4.3.1.2.a would be maintained. The proposed amendment modifies the requirements for new fuel storage configuration for Technical Specification 4.3.1.2.b, to assure the margin of safety is maintained for optimum moderation conditions.

Therefore, operation of WNP–2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: August 20, 1999.

Description of amendment request: The proposed amendment request is to incorporate 17 improvements (identified by Technical Specification Task Force (TSTF) numbers) to the Improved Standard Technical Specifications (TSs), NUREG–1434 (for BWR/6 plants such as the Grand Gulf plant), that was part of the basis for the current improved TSs for Grand Gulf Nuclear Station (GGNS) that were issued in Amendment 120 dated February 21, 1995. These improvements to the improved TSs for BWR/6 plants such as GGNS are identified by TSTF numbers and are the following: (1) TSTF–2, relocate the 10 year sediment cleaning of the diesel generator fuel storage tank in Surveillance Requirement (SR) 3.8.3.6 to the GGNS Updated Final Safety Analysis Report (UFSAR), (2) TSTF–5, delete notification, reporting, and restart requirements if a safety limit is violated in TSs Section 2.2, (3) TSTF–9, relocate the shutdown margin values in Limiting Conditions for Operation (LCO) 3.1.1 and SR 3.1.1.1 to the Core Operating Limits Report (COLR), (4) TSTF–17, extension of the testing frequency for the primary containment airlock interlock mechanism from 184 days to 24 months in SR 3.6.1.2.3 and deletion of the SR Note, (5) TSTF–18, reword and clarify SR 3.6.4.1.2 to require only one secondary containment access door per access opening to be closed, (6) TSTF–32, move the requirement to ensure that ''slow'' and withdrawn stuck control rods are appropriately separated from LCO 3.1.4 requirements to LCO 3.1.3 Condition A Required Actions, (7) TSTF–33, administrative change to clarify the Completion Time for LCO 3.1.3 Required Action A.2, (8) TSTF–38, revise and clarify the visual surveillance in SR 3.8.4.3 for batteries to specify the inspection is for performance degradation, (9) TSTF–45, revise SRs 3.6.1.3.2 and 3.6.1.3.3 to specify that only Primary Containment Isolation Valves which are not locked, sealed, or otherwise secured are required to be verified closed, (10) TSTF–60, exempt LCO 3.4.7 on Reactor Coolant System Leakage Detection Instrumentation from LCO 3.0.4 which restricts entry into MODES, or specified conditions with required equipment inoperable, (11) TSTF–104, relocate the discussion of exceptions in LCO 3.0.4 to the Bases of the TSs, (12) TSTF–118, add a sentence to the administrative controls program in TSs Administrative Controls Section 5.5.9 that the provisions of SRs 3.0.2 and 3.0.3 applies to the specified testing frequencies of the Diesel Fuel Oil Testing Program, (13) TSTF–153, clarify the exception Notes for LCOs 3.4.9, 3.4.10, 3.9.8, and 3.9.9 to be consistent with the requirement being excepted, (14) TSTF–163, modify SRs 3.8.1.2, 3.8.1.12, 3.8.1.15, and 3.8.1.20 for diesel generators to provide minimum volt/Hz limits for the 10-second acceptance and detail the current volt/Hz range as ''steady state'' acceptance criteria, (15) TSTF–166, revise LCO 3.0.6 to explicitly require an evaluation per the Safety Function Determination Program and delete the statement that ''additional * * * limitations may be required,'' (16) TSTF–278, LCO 3.8.6 is revised to require that battery cell parameters be ''within limits,'' the reference to Table 3.8.6–1 is deleted, and a reference to the table is added to the Actions Table for LCO 3.8.6, and (17) TSTF–279, delete the reference to the ''applicable supports'' from the description of the ''Inservice Testing Program'' in the Administrative Controls TSs, Section 5.5.6. The licensee is proposing the current latest revision for each TSTF at the time of application with minor exceptions and/or clarification in some cases.

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 (NSHC). The licensee's NSHC is divided into the following five categories (which also list the TSTF changes in each category): administrative changes, less restrictive changes—removed detail, less restrictive changes—relaxation of required action, less restrictive changes—deletion of surveillance requirement, and less restrictive changes—relaxation of surveillance frequency. The licensee's category NSHCs are presented below:

1. Administrative Changes

These changes involve reformatting, renumbering, and rewording of [TSs], with no change in intent. Since they do not change the intent of the [TSs] they are considered to be administrative in nature. The GGNS is adopting NRC [Nuclear Regulatory Commission] approved TSTF–5, TSTF–18, TSTF–33, TSTF–38, TSTF–104, TSTF–118, TSTF–153, TSTF–163, TSTF–166, TSTF–278, and TSTF–279, generic changes to the Improved Standard Technical Specifications (ISTS) as outlined in NUREG–1434, ''Standard Technical Specifications, BWR/6 Plants.'' In accordance with the criteria set forth in 10 CFR 50.92, EOI [Entergy Operations, Inc.] has evaluated these proposed [TSs] changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change involves reformatting, renumbering, and rewording the existing [TSs]. The reformatting, renumbering, and rewording process involves no changes in intent to the [TSs]. The proposed changes also involve [TSs] requirements, which are purely administrative in nature. As such, this change does not [a]ffect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed change will not reduce a margin of safety because it has no [a]ffect on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

2. Less Restrictive Changes—Removed Detail

GGNS is adopting NRC approved TSTF–2, TSTF–9, and TSTF–32 generic changes to the Improved Standard Technical Specifications (ISTS) as outlined in NUREG–1434, ''Standard Technical Specifications, BWR/6 Plants.'' The proposed changes involve moving details out of the [TSs] and into the [TSs] Bases, the UFSAR, or the Core Operating Limits Report (COLR). The removal of this information is considered to be less restrictive because it is no longer controlled by the [TSs] change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG–1434 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, the EOI has evaluated these proposed [TSs] changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relocates certain details from the [TSs] to other documents under regulatory control. The Bases and UFSAR will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the [TSs] Bases are subject to the change control provisions in the Administrative Controls Chapter of the [TSs]. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). The COLR is controlled in accordance with TS[s] 5.6.5. The controls of TS[s] 5.6.5 will ensure that adequate limits are maintained and reported to the NRC. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no [a]ffect on any safety analysis assumptions. In addition, the details to be moved from the [TSs] to other documents remain the same as the existing [TSs]. Since any future changes to these details will be evaluated, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with the BWR/6 Standard Technical Specifications, NUREG–1434, issued by the NRC Staff, revising the [TSs] to reflect the approved level of detail, which indicates that there is no significant reduction in the margin of safety.

3. Less Restrictive Changes—Relaxation of Required Action

GGNS is adopting NRC approved TSTF–60 generic changes to the Improved Standard Technical Specifications (ISTS) as outlined in NUREG–1434, ''Standard Technical Specifications, BWR/6 Plants.'' The proposed changes involve relaxation of the Required Actions in the current Technical Specifications (TS).

Upon discovery of a failure to meet an LCO, the TS specifies Required Actions to be completed for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from this change is acceptable because the Required Actions take into account the operability status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the EOI has evaluated these proposed [TSs] changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the change have been developed to provide assurance that appropriate remedial actions are taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the change have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the justification, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA [design basis accident] occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

4. Less Restrictive Changes—Deletion of Surveillance Requirement

GGNS is adopting NRC approved TSTF–45 which is a generic change to the Improved Standard Technical Specifications (ISTS) as outlined in NUREG–1434, ''Standard Technical Specifications, BWR/6 Plants.'' The proposed changes involve deletion of [SRs] in the current Technical Specifications (TS).

The TS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. These changes eliminate unnecessary TS [SRs] that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. These changes have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the EOI has evaluated these proposed [TSs] changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change deletes [SRs]. Surveillance's are not initiators 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 [a]ffected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining [SRs] are consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillance's from creating a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deleted [SRs] do not result in a significant reduction in the margin of safety. As provided in the justification, the change has been evaluated to ensure that the deleted [SRs] are not necessary for verification that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

5. Less Restrictive Changes—Relaxation of Surveillance Frequency

GGNS is adopting NRC approved TSTF–17 which is a generic change to the Improved Standard Technical Specifications (ISTS) as outlined in NUREG–1434, ''Standard Technical Specifications, BWR/6 Plants.'' The proposed changes involve the relaxation of Surveillance Frequencies in the current Technical Specifications (TS).

Surveillance Frequencies specify time interval requirements for performing surveillance testing. Increasing the time interval between Surveillance tests results in decreased equipment unavailability due to testing which also increases equipment availability. Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced and[,] in turn, reliability of the [a]ffected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. These changes have been found to be acceptable based on a combination of the above criteria and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the EOI has evaluated these proposed [TSs] changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be Operable and capable of performing any accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly [a]ffected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the justification, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 4, 1999. This amendment request supercedes the licensee's application of June 10, 1999, in its entirety. (64 FR 38025)

Description of amendment request: The proposed amendment would remove the existing filter testing requirements of the Technical Specifications (TSs) and replace them with a reference to the Ventilation Filter Testing Program which is being added to the Administrative Controls section of the Davis-Besse TS. The amendment introduces TS 6.8.4.f, ''Ventilation Filter Testing Program,'' and removes the specific ventilation filter testing requirements from the surveillance requirements of TS 3/4.6.4.4, ''Hydrogen Purge System,'' TS 3/4.6.5.1, ''Shield Building Emergency Ventilation System,'' and TS 3/4.7.6.1, ''Control Room Emergency Ventilation System.'' Also included are supporting Bases changes to TS 3/4.6.4.4, TS 3/4.6.5.1, and TS 3/4.7.6.1

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station(DBNPS), Unit Number 1, in accordance with this change would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. The replacement of the specific Technical Specification (TS) ventilation filter testing Surveillance Requirements for the Containment Hydrogen Purge System (3/4.6.4.4), Shield Building Emergency Ventilation System (3/4.6.5.1), and the Control Room Emergency Ventilation System (3/4.7.6.1), with a reference to the newly created Ventilation Filter Testing Program contained in TS Administrative Controls Section 6.8.4.f, Ventilation Filter Testing Program, is a removal and relocation of certain TS details. The proposed TS 6.8.4.f will, however, add controls to maintain similar operation, maintenance, testing and system operability for these three ventilation systems. The TS Bases changes reflect the use of the Ventilation Filter Testing Program.

The replacement of ASTM D 3803–1979 with ASTM D 3803–1989 for laboratory testing of the charcoal filter samples reflects the NRC recommendations in Generic Letter 99–02, ''Laboratory Testing of Nuclear Grade Activated Charcoal.'' ASTM D 3803–1989 is a more stringent testing standard for charcoal filter testing, than the present standard referenced by the TS.

The increase in allowable charcoal penetration due to the use of a safety factor of ''2'' is acceptable as a result of using this more stringent testing standard.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The increase in allowable charcoal penetration due to the use of a safety factor of ''2'' is acceptable as a result of using this more stringent testing standard. No physical alterations of the DBNPS are involved, nor are plant operating methods being changed. The proposed changes do not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not change the way the plant is operated. No new or different types of failures or accident initiators are being introduced by the proposed changes.

3. Not involve a significant reduction in a margin of safety because there are no significant changes to the initial conditions contributing to accident severity or consequences. Therefore, there are no significant reductions in a margin of safety. Testing under the more restrictive requirements of ASTM D 3803–1989 will continue to ensure that the ventilation systems will perform their safety function.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 1, 1999, as supplemented December 15, 1999.

Description of amendment request: The licensee is requesting to revise the Turkey Point Plant Physical Security Plan (PSP) to modify the PSP requirements for compensation of a security computer failure, and to modify the requirements of the minimum security force staffing. The December 1, 1999, submittal supersedes two previous submittals dated March 10 and June 8, 1999, regarding the same subject. As a result of the proposed changes, License Conditions 3.L. for Turkey Point Units 3 and 4 Operating Licenses will be updated to reflect the latest revision to the Physical Security Plan dated December 1, 1999. In addition, the phrase ''Turkey Point Plant, Units 3 and 4 Security Plan'' was revised to ''Turkey Point Physical Security Plan.'' The latter changes are administrative in nature.

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

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

These changes will not significantly affect the ability to detect a Protected Area intrusion. These changes do not affect the ability of a security response to an overt attack on the plant. These changes will not affect the ability of the security force to respond to contingency events. Therefore, the proposed changes do not affect the probability or consequences of accidents previously analyzed.

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

These changes do not affect the ability of the security force to defeat the design basis threat. The composition of the response organization is not effected by these changes.

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

The demonstrated level of dependability of the security system ensures that a significant reduction in effectiveness or margin of safety does not occur.

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. The staff has also reviewed the changes to License Conditions 3.L. for Turkey Point Units 3 and 4 Operating Licenses, as well as the change of the security plan title. Based on this review, the staff finds that the changes are administrative in nature and that they meet the three criteria discussed above. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: October 6, 1999.

Description of amendment request: This proposed Technical Specification TS change will revise the Cooper Nuclear Station (CNS) TS Sections 1.0, ''Use and Application,'' 3.6, ''Containment Systems,'' Bases 3.0, ''Limiting Condition for Operation (LCO) Applicability,'' Bases 3.6, ''Containment Systems,'' and 5.5, ''Programs and Manuals,'' to adopt the implementation requirements of 10 CFR Part 50, Appendix J, Option B, for the performance of Type A, B, and C containment leakage rate testing. Contingent upon the Nuclear Regulatory Commission's (NRC's) approval of the proposed TS change, the licensee is also requesting the NRC to grant the withdrawal of two exemptions. These exemptions were previously granted under Option A to 10 CFR Part 50, Appendix J; however, under Option B they are no longer required.

The proposed TS change also contains line-item changes for TS requirements addressing containment airlock interlocks, primary and secondary containment isolation valves and power-operated automatic valves. These changes, along with the specific change to implement Option B, have been previously approved by the NRC through submittals made by the Nuclear Energy Institute-sponsored TS Task Force.

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

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

Implement 10 CFR 50 Appendix J, Option B.

There is no increase in the probability or consequences of an accident since there is no work that would affect containment integrity. The testing of containment isolation valves and other containment penetration sealing devices are not postulated as an accident precursor or initiating event.

The NRC has concluded, prior to approving Option B, that performance-based testing would eliminate or modify prescriptive regulatory requirements for which the burden is marginal-to-safety. Reviews and analyses considered by the NRC are presented in NUREG–1493, ''Performance-Based Containment Leak-Test Program, Final Report,'' September 1995 (Attachment 2, Reference 12 [of the October

6, 1999, application]). The historical leakage rate test results for Cooper and for the nuclear industry support extension of the testing frequencies and demonstrate that structural integrity has been maintained.

Type A testing is capable of determining the total leakage from both local leakage paths and gross containment leakage paths. The Type B and C testing has consistently provided accurate leakage rates for valves and penetrations. Administrative controls govern maintenance and testing such that there is very low probability that unacceptable maintenance or alignments can occur. Prior to and following maintenance on primary containment isolation valves and penetrations, a local leak rate test is required to be performed. As a result, Type A testing is not required to accurately quantify the leakage through containment penetrations.

Extension of testing frequency of containment airlock interlock mechanism from 18 months to 24 months.

There is no increase in the probability or consequences of an accident since there is no work that would affect containment integrity. The testing of containment airlock interlocks, isolation valves and other containment penetration sealing devices is not postulated as an accident precursor or initiating event.

This changed the testing of the containment airlock interlocks from 18 months to 24 months. This testing is only performed during periods of reactor shut down and the primary containment is de-inerted. Thus this change plus the allowance from SR [Surveillance Requirement] 3.0.2, provides a total of 30 months, which corresponds to the overall airlock leakage test frequency under Option B. In this fashion, the interlock can be tested in a Mode where the interlock is not required.

Clarify the Containment Isolation Valve (CIV) surveillance to apply to only automatic isolation valves.

The Bases for SR 3.6.1.3.5 state that the isolation time test ensures the valve will isolate in time period less than or equal to that assumed in the safety analysis. There may be valves credited as containment isolation valves, which are power operated, that do not receive a containment isolation signal. These valves do not have an isolation time as assumed in the accident analyses since they require operator action. However, these valves are tested in accordance with the Inservice Test Program as required. Therefore this change reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis.

Based on the above discussion, there is no increase in the probability or consequences of an accident, since this change provides clarification of the applicability of the SR and has no affect on those automatic valves with operating times assumed in the accident analysis.

Allow administrative means of position verification for locked or sealed valves.

It is sufficient to assume that the initial establishment of component status (e.g., isolation valve closed) was performed correctly. Subsequently verification is intended to ensure the component has not been inadvertently repositioned. Given that the function of locking, sealing or securing components is to ensure the same avoidance of inadvertent repositioning, the periodic re-verification should only be a verification of the administrative control that ensures that the component remains in the required state. It would be inappropriate to remove the lock, seal, or other means of securing the component solely to perform an active verification of the required state. There is no increase in the probability or consequences of an accident since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

Therefore, the proposed change described above does not involve a significant increase in the probability or consequences of an accident previously evaluated in the USAR [updated safety analysis report].

The proposed change will not create the possibility of a new or different kind of accident than evaluated in the USAR.

The proposed change involves individual proposed changes related to the implementation of 10 CFR 50 Appendix J, Option B, the extension of testing frequency of the containment airlock interlock, clarification of the CIV surveillance to apply to only automatic isolation valves, and the allowance of administrative means of position verification for locked or sealed valves. The proposed change does not result in any physical change to plant structures, systems, or components. The proposed change does not alter the form, fit, or function of any equipment or components credited in the accident analyses described in the USAR. The performance history of containment testing verifies that containment integrity has been maintained.

The frequency changes allowed by the implementation of the applicable proposed TS changes will not significantly decrease the level of confidence in the ability of the containment to limit offsite doses to allowable values. No accident or malfunction can be the result of the allowed changes to test schedule or frequency.

Since the proposed changes will not directly impact equipment, procedures or operations, the changes will not create the possibility of any new or different kind of accident from any accident previously evaluated in the USAR.

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

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

The reason for performing containment leakage rate testing is to assure that the leakage paths are identified, and that any accident release will be restricted to those paths assumed in the safety analysis. The purpose for the schedule is to assure that containment integrity is verified on a periodic basis. Implementation of Option B to provide flexibility in the scheduled requirements does not mean that containment integrity will be compromised.

The NRC has concluded, prior to approving Option B, that performance-based testing would eliminate or modify prescriptive regulatory requirements for which the burden is marginal-to-safety. Reviews and analyses considered by the NRC are presented in NUREG–1493, ''Performance-Based Containment Leak-Test Program, Final Report,'' September 1995 (Attachment 2, Reference 12). The historical leakage rate test results for CNS and for the nuclear industry support extension of the testing frequencies and demonstrate that structural integrity has been maintained.

Administrative controls govern position verification for locked or sealed valves such that there is a very low probability that unacceptable alignment can occur.

When the containment airlock interlock is opened during times the interlock is required, the operator first verifies that one door is completely shut before attempting to open the other door. Therefore, the interlock is not challenged except during actual testing of the interlock. Therefore, it should be sufficient to ensure proper operation of the interlock by testing the interlock on a 24 month interval.

There may be valves credited as containment isolation valves, which are power operated, that do not receive a containment isolation signal. These valves do not have an isolation time as assumed in the accident analyses since they require operator action. However, these valves are tested in accordance with the Inservice Test Program as required and as such there will be no reduction in a margin of safety.

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: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 6, 1999.

Description of amendment request: Changes are proposed to Technical Specification (TS) Section 2.1.1.2 for the safety limit minimum critical power ratio (SLMCPR). The proposed changes to TS 2.1.1.2 revise the SLMCPR values from 1.06 to 1.08 for two recirculation loop operation, and from 1.07 to 1.09 for single recirculation loop operation.

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

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

Evaluation: The basis for the Safety Limit Minimum Critical Power Ratio (SLMCPR) is to ensure that at least 99.9% of all fuel rods in the core avoid transition boiling if the SLMCPR limit is not violated. The revised SLMCPR values preserve the existing margin to transition boiling and thus the probability for fuel damage is not increased. The determination of a revised SLMCPR Technical Specification value does not affect the assumptions of accidents previously evaluated; or initiate, or affect initiators, of accidents previously evaluated. The proposed revisions to SLMCPR are based on the use of methodology previously accepted by the NRC for calculating SLMCPR and do not change the definition of SLMCPR. Thus, the probability of an accident previously evaluated is not increased.

The revised SLMCPR values do not affect the design or operation of any system, structure, or component in the facility. No new or different type of equipment is installed by this change. The proposed revision does not change or alter the design assumptions for systems, structures, or components used to mitigate the consequences of an accident. Thus, he dose consequences of an accident previously evaluated are not increased.

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

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

Evaluation: The SLMCPR ensures that at least 99.9% of all fuel rods in the core avoid transition boiling if the SLMCPR limit is not violated. The revised SLMCPR values preserve the existing margin to transition boiling. The proposed revisions to SLMCPR are based on the use of methodology previously accepted by the NRC for calculating SLMCPR and do not change the definition of SLMCPR. The proposed revision does not change the design or operation of any system, structure, or component. No new or different type of plant equipment is installed by this change. The proposed revision does not involve a change to plant operation or allowable plant operating modes. The calculational methodology used to determine a revised SLMCPR Technical Specification value cannot initiate or create a new or different type of accident.

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

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

Evaluation: The SLMCPR ensures that at least 99.9% of all fuel rods in the core avoid transition boiling if the SLMCPR limit is not violated. The revised SLMCPR values were calculated using a methodology previously accepted by the NRC, and preserve the existing margin to transition boiling and thus the margin of safety to fuel failure. The proposed change does not involve a relaxation of the criteria or basis used to establish safety limits, or a relaxation in the criteria or bases for the limiting conditions for operation. The assumptions and methodologies used in the plant accident analysis remain unchanged. Therefore, the proposed change does not create a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: December 16, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) Safety Limit Minimum Critical Power Ratio (SLMCPR) values for two recirculation pump and single-loop operation, delete cycle specific footnotes, update the single-loop operation Average Planar Heat Generation rate limiting values, correct a typographical error, and delete an obsolete reference to Siemens fuel.

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

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

GE [General Electric] has recently revised their single loop operation (SLO) analysis review procedures to add an additional requirement that the peak cladding temperature (PCT) during a LOCA [loss-of-coolant accident] initiated while in SLO should be bounded by the PCT for a LOCA initiated while in dual loop operation. This desired result is enforced by revising the SLO MAPLHGR [maximum average planar linear heat generation rate] ''multipliers'' found in Technical Specification 3.11.A from the current value of 0.85 for all fuel to values of 0.78 for GE10 fuel and 0.80 for GE11 and GE12 fuel. This change ensures that the condition that the Upper Bound PCT does not exceed 1600 °F (as required by the NRC-approved SAFER methodology for performing ECCS [emergency core cooling system] LOCA calculations) is satisfied even if a LOCA were to occur while operating in SLO. This change does not alter the method of operating the plant and does not increase the probability of an accident initiating event or transient. These limits are established to preserve required margins.

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

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

SLMCPR is a TS numerical value designed to ensure that transition boiling does not occur in greater than 99.9% of all fuel rods in the core during the limiting postulated transient. A change in SLMCPR cannot create the possibility of any new type of accident. SLMCPR values for the new fuel cycle are calculated using previously transmitted methodology. Similarly, changes to the SLO MAPLHGR multiplier values are designed to ensure that the PCT resulting from a LOCA while operating in SLO are bounded by the PCT from a LOCA while operating in dual loop operation. Thus, a change in these multipliers cannot create the possibility of any new type of accident. This multiplier update results from application of GE's current standard methodology for this analysis.

The proposed changes result only from a specific analysis for the Monticello core reload design and deletion of a cycle specific reference for the values. These changes do not involve any new or different method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes.

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

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

SLMCPR calculations are based on ensuring that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. Proposed SLMCPRs preserve required margin to transition boiling and fuel damage in the event of a postulated transient. Fuel licensing acceptance criteria for SLMCPR calculations apply to Monticello Cycle 20 in the same manner as applied in previous cycles. The revised SLMCPR values do not change the method of operating the plant and have no effect on the probability of an accident-initiating event or transient because these limits are established to preserve required margin.

Fuel licensing acceptance criteria for SLMCPR calculations apply to Monticello Cycle 20 in the same manner as previously applied. SLMCPRs prepared by GE using methodology previously transmitted to the NRC ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle specific transients are evaluated.

Application of new SLO MAPLHGR multiplier values ensures that SLO LOCA results are bounded by those for dual loop operation and thus maintain or improve the margin of safety for LOCA analyses.

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

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

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

NRC Section Chief: Claudia M. Craig.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: August 5, 1999.

Description of amendment request: The proposed amendment would add a license condition denoting NRC approval of the Trojan Nuclear Plant (TNP) License Termination Plan.

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

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

The requested license amendment does not authorize additional plant activities beyond those that already may be conducted under the approved TNP Decommissioning Plan and the Defueled Safety Analysis Report (DSAR). Accident analyses are included in the approved TNP Decommissioning Plan and incorporated into the TNP DSAR. No systems, structures, or components that could initiate or be required to mitigate the consequences of an accident are affected by the proposed change in any way not previously evaluated in the approved TNP Decommissioning Plan and DSAR. Therefore, the proposed change is administrative in nature and as such does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The requested license amendment does not authorize additional plant activities beyond those that already may be conducted under the approved TNP Decommissioning Plan and the DSAR. Accident analyses are included in the approved TNP Decommissioning Plan and incorporated into the DSAR. The proposed change does not affect plant systems, structures, or components in any way not previously evaluated in the approved TNP Decommissioning Plan and DSAR, and no new or different failure modes will be created. Therefore, the proposed change is administrative in nature and as such does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Approval of the TNP License Termination Plan by license amendment is administrative in nature since the decommissioning and fuel storage activities described in the TNP license Termination Plan are consistent with those in the approved TNP Decommissioning Plan and DSAR. 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: Leonard A. Girard, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Michael T. Masnik.

Power Authority of the State of New York, Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: November 24, 1999.

Description of amendment request: The proposed amendment would delete Section 4.7.D.1.e of Appendix A (Technical Specifications (TSs)) to the James A. FitzPatrick Operating License to eliminate the surveillance requirement for partially stroking of the plant Main Steam Isolation Valves (MSIVs) twice a week. The MSIVs will continue to be fully stroked with a frequency that is in accordance with the In-Service Testing (IST) Program per TS 4.7.D.1.d, which is consistent with the Boiling-Water Reactor Standard Technical Specification and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The proposed changes include associated administrative changes to Section 4.7.D.1.d, and to Bases Section 4.7.D of the TSs.

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

(1) The proposed change will not significant[ly] increase the probability or consequences of any previously evaluated accidents.

This proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. This proposed change deletes the requirement to exercise the MSIVs twice per week. The twice per week exercise involves partial closure of each individual MSIV and subsequent reopening to the full open position.

The safety function of the MSIV is to isolate the main steam line in case of a steam line break, Control Rod Drop Accident or Loss of Coolant Accident in order to limit the loss of reactor coolant and/or the release of radioactive materials. The MSIVs perform a safety function which mitigates the consequences of accidents: however, an event can be initiated by the inadvertent closure of MSIVs. Therefore, eliminating excessive operation of the MSIVs reduces the probability of an inadvertent closure. Also, the surveillance which is being deleted does not test the safety function of the MSIVs. The safety function is tested during the full stroke fast closure test. Since deleting the twice per week exercise of the valves is not considered to have any effect on the reliability of the MSIVs to perform there safety function, there is no increase in the consequences of any postulated accidents. Therefore, deleting the requirement for twice per week exercising of the MSIVs does not significantly increase the probability or consequences of any previously evaluated accidents.

(2) The proposed change will not create the possibility of a new or different kind of accident.

The safety function of the MSIV is to isolate the main steam line in case of a steam line break, Control Rod Drop Accident, or Loss of Coolant Accident in order to limit the loss of reactor coolant and/or the release of radioactive materials. The MSIVs perform a safety function which mitigates the consequences of accidents: however, an event can be initiated by the inadvertent closure of MSIVs. The inadvertent closure of the MSIVs event has been previously evaluated in Chapter 14 of the James A. FitzPatrick Final Safety Evaluation Report (FSAR). The surveillance which is being deleted does not test the safety function of the MSIVs. The safety function is tested during the full stroke fast closure test. Since the MSIVs perform a mitigating safety function, and the MSIV full stroke fast closure test adequately tests the safety function, elimination of the twice per week exercise will not create any new or different kind of accident.

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

The safety function of the MSIV is not tested during the twice per week exercise. The ability of the MSIVs to perform their safety function is tested during the MSIV full stroke fast closure test in accordance with the IST Program. Therefore, deletion of the requirement does not reduce the margin of safety. The exercising of the MSIVs was originally specified in order to detect binding of the pilot valve. The type of pilot valve that was susceptible to binding was replaced and there is no longer any need for frequent exercising of the MSIVs. The full closure test of the MSIVs in accordance with the IST Program adequately demonstrates that the MSIVs and their pilot valves are not binding and that the MSIVs will perform their safety function. Additionally, reducing the frequency of MSIV operation reduces the probability of inadvertent scrams and transients, and challenges to relief valves, providing a net addition to the margin of safety. The full stroke fast closure test is considered to be sufficient. It is the only test required by the ASME Boiler and Pressure Vessel Code and the BWR Standard Technical Specifications. Therefore, eliminating the twice per week exercise of the MSIVs does not significantly reduce any margin of safety.

The proposed change will not increase the probability or consequences of any previously analyzed accident, introduce any new or different kind of accident previously evaluated, or reduce existing margin to safety. Therefore, the proposed license amendment will 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: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Alexander W. Dromerick (Acting Section Chief).

Public Service Electric & Gas Company, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: November 24, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to implement Filtration, Recirculation, and Ventilation System (FRVS) and Control Room Emergency Filtration (CREF) System charcoal filter testing requirements that are consistent with the U. S. Nuclear Regulatory Commission requirements delineated in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

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 TS change does not involve any physical changes to plant structures, systems or components (SSC). The CREF and FRVS systems will continue to function as designed. The CREF and FRVS systems are designed to mitigate the consequences of an accident, and therefore, can not contribute to the initiation of any accident. The proposed TS surveillance requirement changes implement testing methods that more appropriately demonstrate charcoal filter capability and establish acceptance criteria, which ensure that Hope Creek's licensing and design basis assumptions are met.

In addition, this proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety, since the manner in which the CREF and FRVS systems are operated is not affected by these proposed changes. The proposed surveillance requirement acceptance criteria ensure that the FRVS and CREF safety functions will be accomplished. Therefore, the proposed TS changes would not result in the increase of the consequences of an accident previously evaluated, nor do they involve an increase in the probability of an accident previously evaluated.

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

The proposed TS changes do not involve any physical changes to the design of any plant SSC. The design and operation of the CREF and FRVS systems are not changed from that currently described in Hope Creek's licensing basis. The CREF and FRVS systems will continue to function as designed to mitigate the consequences of an accident. Implementing the proposed charcoal filter testing methods and acceptance criteria does not result in plant operation in a configuration that would create a different type of malfunction to the CREF and FRVS systems than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in Hope Creek's licensing basis regarding the safety related functions of these systems.

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

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

The proposed changes contained in this submittal would implement TS requirements that: (1) Are consistent with the requirements delineated in Generic Letter 99–02; (2) implement testing methods that adequately demonstrate charcoal filter capability; and (3) establish acceptance criteria consistent with Hope Creek's licensing basis. The ability of CREF and FRVS to perform their safety functions is not adversely affected by these proposed changes. Therefore, the proposed TS 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 California Edison Company, et al., Docket Nos. 50–206, 50–361, and 50–362, San Onofre Nuclear Generating Station, Units 1, 2, and 3, San Diego County, California

Date of amendment requests: December 2, 1999 (Unit 1—PCN 267, Units 2 and 3—PCN 506).

Description of amendment requests: This amendment application is a request to revise the Unit 1 Technical Specifications Section D6, Administrative Controls, to be consistent with the San Onofre Units 2 and 3 Technical Specification Section 5.0, Administrative Controls, and incorporate changes related to certified fuel handlers and 10 CFR 50.54(x), administrative control of working hours and working hour deviation approvals, position titles and responsibilities and organizational description reference, qualifications for a multi-discipline supervisor, quality assurance program control of review and audit and record retention procedures, high radiation area controls, description of the plant configuration for environmental protection, and environmental protection related document reporting.

This amendment application also requests to revise the Unit 2 and Unit 3 Technical Specifications, Section 5.0, Administrative Controls, to incorporate changes related to the operating organization, working hours deviation approvals, qualifications for a multi-discipline supervisor, the schedule for submitting Technical Specification Bases changes, reference to American Society of Mechanical Engineers (ASME) code class components, steam generator inspection reporting, Core Operating Limits Report references, high radiation area controls, offsite dose calculation manual change control reference, and environmental protection related document reporting.

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

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

No. This proposed change is to revise the administrative controls section of the San Onofre Units 1, 2 and 3 technical specifications. To the extent practicable, the San Onofre Unit 1 technical specification Section D6, Administrative Controls, is made consistent with the San Onofre Units 2 and 3 technical specification Section 5.0, Administrative Controls. This change allows the handling of key administrative controls to be consistent on site. Certain position titles have been revised, and the cognizant Vice President has been included as an approver of deviations from the work hours and reviewer of overtime hours. The Vice President—Business and Financial Services is identified to be responsible for Unit 1 decommissioning. The specification allowing the certified fuel handlers to implement 10 CFR 50.54(x) is removed since this is now included in the regulations. The qualification requirements for a multi-discipline supervisor consistent with the American National Standards Institute [ANSI] standard have been added to the staff qualifications section. The schedule for submitting technical specification Bases changes is revised to be consistent with the NRC approved exemption to 10 CFR 50.71(e) for submitting Updated Final Safety Analysis Report (UFSAR) updates. A reference to Class 1, 2, and 3 ASME code components is removed from the technical specifications and maintained in the Licensee Controlled Specifications (LCS) and the inservice inspection and testing program. The Units 2 and 3 steam generator inspection reporting requirements are revised to refer to the technical specification requirement. The Core Operating Limits Report (COLR) section is revised to include references to 2 topical reports related to the reload analysis technology transfer and the NRC's evaluation of the technology transfer. The sections on high radiation are revised to be consistent with Regulatory Guide 8.38 which provides an acceptable method for controlling access to high radiation areas. The environmental protection section of the San Onofre Unit 1 technical specifications is revised to reflect the current status of the discharge system. The environmental protection sections for Unit 1 and Units 2/3 are further revised by including a 30 day timeframe for providing the NRC copies of reports related to unusual or important environmental events and deleting the requirement to provide the NRC copies of proposed changes and renewal applications for NPDES permits.

All of these changes are being made to provide consistency and flexibility in the handling of site programs, and update and clarify the administrative controls. There are no equipment changes or modifications to the plant associated with these changes that would affect the probability or consequences of accidents at all three units.

Therefore, this change does not affect the probability or consequences of an accident previously evaluated.

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

No. This proposed change is to revise the administrative controls sections of the San Onofre Units 1, 2, and 3 technical specifications. The changes provide consistency and flexibility in the handling of site programs, and update and clarify the administrative controls. There is no administrative change being made that could create a new or different accident at any of the three units and there is no plant or equipment modification associated with this change.

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

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

No. This change revises the administrative controls sections of the San Onofre Units 1, 2, and 3 technical specifications. The changes provide consistency and flexibility in the handling of site programs, and update and clarify the administrative controls. There is no change to plant equipment associated with this change. This change does not affect any margin of safety.

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

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chiefs: Michael Masnik (Unit 1); Stephen Dembek (Units 2 and 3).

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

Date of amendment requests: November 24, 1999 (PCN–274).

Description of amendment requests: The licensee proposes to revise Technical Specification (TS) 3.3.11, ''Post Accident Monitoring Instrumentation (PAMI).'' Specifically, the proposed change would extend the PAMI channel calibration surveillance frequency from 18 months to 24 months to accommodate a 24-month fuel cycle. All PAMI instruments would then be on a 24-month calibration interval, which removes the need for Surveillance Requirement (SR) 3.3.11.5. Therefore, the licensee also proposes to delete SR 3.3.11.5.

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

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

Response: No.

The proposed license amendment[s] to extend the calibration surveillance frequency of Post Accident Monitoring Instrumentation (PAMI) instrumentation [are] being made to support plant operation with a 24-month fuel cycle.

Increasing the calibration intervals for PAMI instrumentation to 30 months [24 months plus the 25% surveillance interval extension allowed by SR 3.0.2] does not affect the initiation or probability of any previously analyzed accident. Increasing the calibration interval will not affect the integrity of any of the principal barriers against radiation release (fuel cladding, reactor vessel, and containment building). The ability of the plant to mitigate the consequences of any previously analyzed accidents is not adversely affected.

PAMI instrumentation provides to the operators both qualitative and quantitative information used in accident mitigation and for the safe shutdown of the plant. Instrumentation which provides qualitative information is unaffected by a change in instrument accuracy induced by drift due to the increased surveillance interval because no explicit value is required by the Emergency Operating Instructions (EOIs). Instrumentation that provides quantitative information (*i.e.,* decision points) in the EOIs have been evaluated. This evaluation resulted in no changes to any operating instructions. This evaluation of the proposed change to the surveillance interval demonstrates that licensing basis safety analyses acceptance criteria and San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 EOI criteria will continue to be met.

The proposed new surveillance frequency for these instrument channels was evaluated using the guidance of Generic Letter 91–04. The basis for the change includes a quantitative evaluation of instrument drift for PAMI instrumentation providing quantitative information to the EOIs. Also, loop accuracy/setpoint calculations for these instruments were updated to accommodate the extended surveillance period. Analyses and evaluations completed to assess the proposed increase in the surveillance interval demonstrate that the effectiveness of these instruments in fulfilling their respective functions is maintained. Technical Specifications Channel Checks and Channel Functional Checks for the subject channels, will continue to be performed to provide assurance of instrument channel OPERABILITY.

Therefore, the proposed amendment[s do] not involve a significant increase in the probability or consequences of any previously analyzed accident.

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

Response: No.

The increased calibration surveillance interval for PAMI instrumentation is justified based on evaluation of past equipment performance and does not require any plant hardware changes or changes in normal system operation. Changing the calibration interval for this instrumentation has no means of creating the possibility of a new or different kind of accident. There are no new decision points or operator responses required to support existing accident mitigation strategies.

Therefore, there are no new failure modes introduced as a result of extending these surveillance intervals, and the proposed amendment[s do] not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to the calibration surveillance interval was evaluated using the criteria of 95% probability/95% confidence level for process sensor drift.

PAMI instrumentation are used to provide indication following certain hypothetical accident conditions and are used in EOIs for trending and to initiate operator action at certain decision points. Instrument uncertainty calculations have been updated for PAMI instrumentation used for EOI decision points as appropriate. Updated calculations show that the total loop uncertainty for PAMI evaluated either decreased or remained the same. These updated calculations demonstrate that applicable accuracy requirements for SONGS 2 and 3 are satisfied with the proposed new surveillance intervals.

Changing the calibration interval for these channels does not affect the margin of safety for previously analyzed accidents. Therefore, the proposed amendment[s do] not involve a significant reduction in a margin of safety.

Based on the responses to these three criteria, Southern California Edison (SCE) has concluded that the proposed amendment[s involve] no 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 requests involve no significant hazards consideration.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of amendment requests: December 13, 1999 (PCN–507).

Description of amendment requests: San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 are currently licensed for operation for 40 years commencing with issuance of their construction permits. The licensee proposes to amend the SONGS Units 2 and 3 operating licenses to revise the expiration dates of these licenses to 40 years from the date of issuance of the operating licenses. Thus, these amendment applications request that the SONGS Unit 2 operating license expiration date be changed from October 18, 2013, to February 16, 2022, and the SONGS Unit 3 operating license expiration date be changed from October 18, 2013, to November 15, 2022.

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

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

Response: The proposed change does not involve any changes to the design or operation of the San Onofre Nuclear Generating Station (SONGS) 2 and 3 which may affect the probability or consequences of an accident evaluated in the Updated Final Safety Analysis Report (UFSAR). SONGS 2 and 3 were designed and constructed on the basis of a forty (40) year life. The accidents analyzed in the UFSAR were postulated on the basis of a 40 year life. No changes will be made that could alter the design, construction, or postulated scenarios regarding accident initiation and/or response. Existing surveillance, inspection, testing and maintenance practices and procedures ensure that degradation in plant equipment, structures, and components will be identified and corrected throughout the life of the plant. The effect of aging of electrical equipment, in accordance with 10 CFR50.49, has been incorporated into the plant maintenance and surveillance procedures. Therefore, the probability or consequences of a postulated accident previously evaluated in the UFSAR are not increased as a result of the proposed change.

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

Response: The proposed change does not involve any changes to the physical structures, components, or systems of SONGS 2 and 3. Existing surveillance, inspection, testing, and maintenance practices and procedures will assure full operability for the plant's design lifetime of 40 years. Continued operation of SONGS 2 and 3 in accordance with these approved procedures and practices will not create a new or different kind of accident.

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

Response: There are no changes in the design, design basis, or operation of SONGS 2 and 3 associated with the proposed change. Existing surveillance, inspection, testing, and maintenance practices and procedures provide assurance that any degradation of equipment, structures, or components will be identified and corrected throughout the lifetime of the plant. These measures together with the continued operation of SONGS 2 and 3 in accordance with the Technical Specifications assure an adequate margin of safety is preserved on a continuous basis. Therefore, the proposed change does not result in a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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: November 30, 1999.

Description of amendment request: The proposed amendments would change Technical Specification Surveillance Requirement 3.8.1.12 to remove the restriction which prevents performance of the diesel generator 24-hour run while operating in either Mode 1 or Mode 2.

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

1. The proposed change does not involve a significant increase in the probability or the consequences of a previously evaluated event for the following reasons:

The primary function of the diesel generators is to supply emergency power to the safety-related equipment necessary to safely shut down the plant in case of a design basis event, such as a loss of coolant accident (LOCA) concurrent with a loss of offsite power (LOSP). The diesels are not designed to prevent such an event. Accordingly, the probability of a LOCA/LOSP event is not increased by allowing the performance of the 24-hour run with the reactor operating.

It is possible that, with a diesel generator connected to its bus, an electrical disturbance will travel through the system and affect the other busses. This is most likely to happen when initially connecting the diesel to the bus. However, the surveillance procedures require that diesel generator output voltage be synchronized with the bus prior to the diesel output breaker being closed in, thus reducing the chance of an electrical distribution system disturbance.

If a LOCA occurred concurrent with an LOSP while a diesel generator is connected to the bus in its 24-hour run, the diesel logic automatically realigns itself to the Standby mode of operation, allowing the diesel to supply power to the emergency bus. A Technical Specifications surveillance requirement tests this feature. Also, the proposed specification prevents the test from being performed unless the other two diesel generators are operable; this includes suspending the surveillance if one of the other available diesels becomes inoperable during the actual test. This restriction will ensure that two diesels are available to safely shut down the plant if necessary.

Additionally, this amendment request does not affect any other system or piece of equipment necessary to prevent or mitigate the consequences of previously evaluated events. As a result, the consequences of a LOCA/LOSP event are not increased.

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

This proposed modification to SR 3.8.1.12 does not introduce any new modes of operation or testing. In fact, each diesel generator is already connected to its respective bus during operation to satisfy SR 3.8.1.2, the monthly test. In the monthly test, the diesel is run loaded for 1 hour, connected to the grid, with the unit in operation. Therefore, allowing the 24 hour test to be performed for the diesels introduces nothing new with respect to diesel testing, and as a result, the possibility of a new type of event is not created.

3. The change does not significantly reduce the margin of safety for the following reasons:

The probability of an electrical disturbance affecting plant operation while connecting the diesel to the bus is minimized by the fact that the diesel's output voltage and phase angle are synchronized with those of the grid prior to being tied to the emergency bus. Based on engineering judgement, with the diesel synchronized and running connected to the grid, the likelihood of an electrical disturbance being transferred through the system and causing a plant transient is very small. Furthermore, since only one diesel will be tied to the bus in either Mode 1 or Mode 2, neither of the other two diesel generators will be affected by the disturbance.

If a LOCA/LOSP occurred during the 24-hour run, the diesel generator's auto-logic would take the diesel out of the test mode. This feature is tested once per 18 months per Technical Specifications. With the diesel no longer in test, it would be free to once again tie itself to the bus. Additionally, only one diesel will be tied to the line during a 24-run performed with the reactor operating, with other diesel generators available to supply power to their respective emergency busses. This ensures two diesels are available to shut down the plant and maintain it in a safe condition.

Other precautions will also be placed into plant procedures; specifically, the 24-hour run will not be performed on line during periods of severe weather or during grid instabilities.

For the above reasons, the proposed Technical Specifications change will not significantly reduce the margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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 18, 1999.

Description of amendment request: The proposed amendments would revise technical specification surveillance requirements 4.7.7, 4.7.8, and 4.9.12, on the control room makeup and cleanup filtration system and the fuel handling building exhaust air system, from a requirement that laboratory analysis of charcoal filter samples meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, ''Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants,'' Revision 2, March 1978, to a requirement that the analysis meets the laboratory testing criteria of American Society for Testing and Materials ASTM D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.''

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

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

The proposed change revises the test protocol for Engineered Safety Feature charcoal filters from ASTM D3803–1979 to ASTM D3803–1989. The change in protocol is a conservative change in that the revised test conditions will more accurately reflect the functionality of the charcoal filters under accident conditions. There is no change in plant configuration or components. The tests are conducted under laboratory conditions, so that change in protocol has no effect on plant operation. There is no change in how samples are taken to be used in analyses.

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

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

The proposed change revises the test protocol for Engineered Safety Feature charcoal filters from ASTM D3803–1979 to ASTM D3803–1989. The change in protocol is a conservative change in that the revised test conditions will more accurately reflect the functionality of the charcoal filters under accident conditions. There is no change in plant configuration or components. The tests are conducted under laboratory conditions, so that change in protocol has no effect on plant operation. There is no change in how samples are taken to be used in analyses.

Based on the above, this 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 a significant reduction in a margin of safety.

The proposed change revises the test protocol for Engineered Safety Feature charcoal filters from ASTM D3803–1979 to ASTM D3803–1989. The change in protocol is a conservative change in that the revised test conditions will more accurately reflect the functionality of the charcoal filters under accident conditions. There is no change in plant configuration or components. The tests are conducted under laboratory conditions, so that change in protocol has no effect on plant operation. There is no change in how samples are taken to be used in analyses.

Based on the above, the margin of safety is not significantly reduced by this change.

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: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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: December 6, 1999.

Description of amendment request: The proposed amendments would revise Technical Specification Definition 1.9, ''Core Alterations,'' to explicitly define core alterations as the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel.

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

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

The proposed change does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed change does not involve any physical changes to the facility. The change to the definition of core alterations is consistent with that used in NUREG–1431, Revision 1, ''Improved Standard Technical Specifications for Westinghouse Plants.'' The proposed revision to the definition of core alterations will not affect the Technical Specifications Section 3/4.9, ''Refueling Operations'', requirements which ensure the core remains subcritical, nor will any Limiting Condition for Operation required for core alterations or the movement of fuel be changed. The proposed change will not affect any safety margin or safety limit applicable to the facility. Therefore, the proposed change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not affect any previously evaluated accident scenario, nor does it create any new accident scenarios. The proposed change is a clarifying revision to the definition of core alterations only, and will not alter any of the currently approved refueling operation activities, nor will it create any new refueling operation activities.

Since the proposed change does not impact operation of the facility as presently approved, no possibility exists for a new or different kind of accident from those previously evaluated.

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

South Texas Project Technical Specification 3/4.9.1, ''Boron Concentration'', ensures that the reactor will remain subcritical ($K_{eff} \leq 0.95$) during core alterations and that uniform boron concentration is maintained for reactivity control in the water volume having direct access with the reactor vessel. The proposed change in the definition of core alterations will allow ''non-reactive'' components, such as cameras, lights, tools, movable incore detector thimbles, etc., to be moved or manipulated in the vessel, with fuel in the vessel and the vessel head removed, without constituting a core alteration. This is acceptable because these types of components will have negligible effect on core reactivity, and will not affect reactor coolant system boron concentration. Therefore, operations using these types of components will not adversely affect K_{eff} or the shutdown margin. Additionally, reactor subcriticality status is continuously monitored in the control room during Operating Mode 6, as specified in Specification 3/4.9.2, ''Instrumentation''. Thus, there will be no reduction in a margin of safety resulting from the proposed change.

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: Jack R. Newman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

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 application for amendments: October 14, 1999 (TS 99–12).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Operating Licenses DPR–77 (Unit 1) and DPR–79 (Unit 2) by revising the Technical Specification (TS) surveillance requirements for steam generator tube integrity by incorporating an alternate repair criteria for axial primary water stress corrosion cracking at dented tube support plate intersections.

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

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

Operation of Sequoyah Units 1 and 2, in accordance with the proposed license amendment, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Examination of crack morphology for primary water stress corrosion cracking (PWSCC) at dented intersections has been found to show one or two microcracks well aligned with only a few uncorroded ligaments and little or no other inside diameter axial cracking at the intersection. This relatively simple morphology is conducive to obtaining good accuracy in Non-destructive Examination (NDE) sizing of these indications. Accordingly, alternate repair criteria is established based on crack length and average and maximum depth within the thickness of the tube support plate (TSP) or limited extension outside the thickness of the TSP.

The application of the alternate repair criteria (ARC) requires a condition monitoring assessment. If all indications satisfy the structural limits with regard to bounding lengths and average depths, the condition monitoring burst pressure requirements are satisfied.

In addition, an operational assessment is performed to determine the length/depth repair bases. The crack profiles are projected to the end of the operating cycle for comparison with acceptance limits (i.e., length limit and average depth limit). Burst pressures are calculated from the depth profiles by searching the total crack length for the partial length that results in the lowest burst pressure. Because the burst pressure can be lower than that for the longest acceptable crack length at its average depth, a fixed repair limit is not established. The repair bases is obtained by projecting the crack profile to the end of the next operating cycle and determining if the burst pressure for the projected profile meets the burst pressure margin requirements defined by [Westinghouse Topical Report] WCAP–15128, Revision 1, dated August 1999. If the projected end-of-cycle (EOC) burst margin requirements are satisfied, the indication is left in service. Thus, the repair limit relative to length and average depth assures that the operational assessment requirements are satisfied.

Crack length limits are established in the WCAP to assure that crack extension and growth outside of the TSP provides adequate margin against burst for the free-span crack (*i.e.,* $3DP_{NO}$ burst capability is maintained) in addition to the total crack length. A repair limit is also established in the WCAP for maximum depth to provide a high confidence that the indication will not progress through the wall at the end of an operating cycle.

Based on the above, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated within the Sequoyah FSAR [Final Safety Analysis Report].

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

Implementation of the proposed S/G [steam generator] tube ARC does not introduce any significant changes to the plant design basis. A single or multiple tube rupture event would not be expected in a S/G in which the plugging criteria has been applied. Both condition monitoring and operational assessments are completed as part of the implementation of ARC to determine that structural and leakage margin exists prior to returning S/Gs to service following inspections. If the condition monitoring requirements are not satisfied for burst or leakage, the causal factors for EOC indications exceeding the expected values will be evaluated. The methodology and application of this ARC will continue to ensure that tube integrity is maintained during all plant conditions consistent with the requirements of draft RG [Regulatory Guide] 1.121 and Revision 1 of RG 1.83.

A S/G tube rupture event is one of a number of design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a S/G tube rupture event, a bounding primary-to-secondary leakage rate equal to the operational leakage limits in the TSs, plus the leak rate associated with the double ended rupture of a single tube, is assumed. For other design basis accidents such as a main steam line break and loss of alternating current power, the tubes are assumed to retain their structural integrity and exhibit primary-to-secondary leakage within the limits assumed in Final Safety Analysis Report (FSAR) accident analyses. The proposed ARC does not result in an accident leakage rate in excess of that assumed or calculated in SQN's current accident analyses.

Even under severe accident conditions, the potential for significant leakage would be expected to be small and not significantly different than for other degradation mechanisms repaired to 40 percent depth limits. It is concluded that application of the proposed ARC for PWSCC at dented TSP locations results in a negligible difference from current 40-percent repair limits.

TVA continues to implement a maximum operating condition leak rate limit of 150 gallons per day (0.1 gallons per minute) per S/G to preclude the potential for excessive leakage during all plant conditions.

The possibility of a new or different kind of accident from any previously evaluated is not created because S/G tube integrity is maintained by inservice inspection and effective primary-to-secondary leakage monitoring.

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

Tube repair limits provide reasonable assurance that tubes accepted for continued service without plugging or repair will exhibit adequate tube structural and leakage integrity during subsequent plant operation. The implementation of the proposed ARC is demonstrated to maintain S/G tube integrity consistent with the criteria of draft NRC Regulatory Guide 1.121. The guidelines of RG 1.121 describe a method acceptable to the NRC staff for meeting General Design Criteria (GDC) 2, 4, 14, 15, 31, and 32 by ensuring the probability or the consequences of S/G tube rupture remain within acceptable limits. This is accomplished by determining the limiting conditions of degradation of S/G tubing, for which tubes with unacceptable cracking should be removed from service.

Upon implementation of the proposed ARC, even under the worst-case conditions, the occurrence of PWSCC at the tube support plate elevations is not expected to lead to a S/G rupture event during normal or faulted plant conditions. All tubes are shown to retain the margins of safety against burst consistent with the safety factor margins implicit in the stress limit criteria of Section III of the American Society of Mechanical Engineers [ASME] Code, for all service loading conditions. In addition, all tubes have been shown to retain a margin of safety against gross failure or burst consistent with the stress limits of [Paragraph] NB–3225 of Section III of the ASME Code under postulated accident conditions concurrent with a safe shutdown earthquake.

In addressing the combined effects of loss-of-coolant accident plus safe shutdown earthquake on the S/G component (as required by GDC 2), it has been determined that tube collapse will not occur in the Sequoyah S/Gs. This analysis is discussed in WCAP 13990, dated May 1994. No tubes are excluded from the application of the proposed ARC.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to the plant safety analyses as defined in the FSAR or TSs.

The NRC 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 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: November 8, 1999.

Brief description of amendments: The proposed amendments would change Technical Specification 5.5.11, ''Ventilation Filter Testing Program (VFTP)'' to include the requirement for laboratory testing of Engineered Safety Feature (ESF) Ventilation System charcoal samples per American Society for Testing and Materials (ASTM) D3803–1989 and the application of a safety factor of 2.0 to the charcoal filter efficiency assumed in the plant design-basis dose analyses.

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

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

The proposed changes only involve the laboratory testing methodology performed on activated charcoal to help determine whether the charcoal in the filtration units can remain in place or [if it] require[s] replacement.

Generic Letter 99–02 intends to standardize the way nuclear-grade activated charcoal is tested throughout the industry in order to provide conservative filtration results as well as uniform and repeatable tests. The purpose is to ensure the filtration systems protect the Operators in the Control Room (GDC [General Design Criterion] 19) as well as the public (10CFR100), in the event of a radiological accident scenario.

The charcoal adsorber sample laboratory testing per ASTM D3803–1989 is more stringent than the current testing practice and more accurately demonstrates the required performance of the adsorbers following a design ba[s]is LOCA [loss of coolant accident]. No Licensing Basis Accidents or mitigation capability will be affected by incorporation of these changes.

Therefore, this change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

Plant procedures are only altered to the extent that the revised specification will allow different reference standards for testing activated charcoal. These changes ensure continued support of the safety related ESF filtration equipment and do not affect their failure or failure modes.

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

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

None of the changes being proposed alter the environmental conditions maintained in the areas supported by the ESF filtration systems during normal operations and following an accident. Also these changes will not cause an increase in radiological releases through the Primary Plant Ventilation Exhaust System. As a result, the margin of safety for these functions remains the same.

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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of application request: December 3, 1999 (ULNRC–04158).

Description of amendment request: The proposed amendment requested changes to Section 5.6.6, ''Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR),'' of the improved Technical Specifications (ITS) that were issued on May 28, 1999, in Amendment No. 133. The current Technical Specifications (CTS) remain in effect until the ITS are implemented on or before April 30, 2000. The proposed changes to the ITS would approve the use of the PTLR by the licensee to make changes to the plant pressure temperature limits and low temperature overpressure protection limits without prior NRC staff approval in accordance with Generic Letter 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,'' dated January 31, 1996. The proposed changes are: (1) Add the word criticality to ITS Subsection 5.6.6.a as one of the reactor conditions for which RCS pressure and temperature limits will be determined, (2) add the phrase ''and COMS PORV,'' where COMS PORV stands for cold overpressure mitigation system power operated relief valve, to the the introductory paragraph of ITS subsection 5.6.6.b to show that the analytical methods listed in the subsection are also for the COMS PORV, and (3) replace the two documents listed in ITS subsection 5.6.6.b by the reference to the future NRC letter that approves the use of the PTLR and the Westinghouse Topical Report, WCAP–14040–NP–A, Revision 2, ''Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'' dated January 1996, that provides the methodology that will be used by the licensee in using the PTLR report. The current plant pressure temperature limits and low temperature overpressure protection limits are in the CTS and were approved in Amendment No. 124, which was issued April 2, 1998.

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

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

The proposed change submits the PTLR, which contains the relocated CTS heatup and cooldown, and COMS PORV limits and the methodology used to calculate them, and the added references into ITS 5.6.6. The proposed change is administrative in nature since it is a movement of information from the CTS to a licensee controlled document, and has prior NRC staff approval. The PTLR contains the limit curves and the ITS requires more restrictive actions to be taken when the limiting conditions for operation are not met than is currently required by the CTS. The heatup and cooldown, and COMS PORV limits within the PTLR will be implemented and controlled per Callaway Plant programs and procedures and changes to the PTLR will be performed per requirements of 10 CFR 50.59 to ensure that change to these limits in the future will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As stated earlier, the movement of the heatup and cooldown, and COMS PORV limits from the CTS to the PTLR has no influence or impact, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operations will be altered as a result of this proposed change. The proposed change is administrative in nature since it is a movement of requirements from the CTS to a licensee controlled document, the PTLR, and the change adds references into the ITS incorporating the licensee controlled document. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change does not affect the acceptance criteria for an analyzed event. The margin of safety presently provided by the CTS remains unchanged. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protective functions. Therefore, the proposed change is administrative in nature and does not impact the operation of Callaway Plant in a manner that involves a reduction in a margin of safety.

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: November 5, 1999, as supplemented on December 3, 1999.

Description of amendment request: This proposed change revises the applicability for the reactor power distribution limits and the Average Power Range Monitor (APRM) gain adjustments. The applicability is proposed to be revised to operation at ≥25% Rated Thermal Power (RTP).

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

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

The proposed change does not involve an increase in the probability or consequences of an accident previously evaluated because the revisions standardize and make consistent the applicability and actions for the reactor power distribution limits in the current Technical Specifications. Since reactor operation with these revised Specifications is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events or assumed mitigation of accident or transient events. The structural and functional integrity of plant systems is unaffected. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not affect any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created. No safety-related equipment or safety functions are altered as a result of these changes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

At thermal power levels < 25% RTP, the reactor is operating with substantial margin to the reactor power distribution limits [and this margin is unchanged]. The proposed change does not impact operation at power levels ≥ 25% RTP and has no effect on any safety analysis assumption or initial condition. Thus, the margin of safety required for safety analyses [is] maintained. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Wisconsin Electric Power Company, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant (PBNP), Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: November 15, 1999 (TSCR 202).

Description of amendment request: The proposed amendments would change the Technical Specifications (TSs) in order to extend the required frequency of the control rod exercise test (TS 15.4.1, Table 15.4.1–2, Item 10) from the current frequency of every 2 weeks to quarterly.

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

Relaxing the frequency of performance for a surveillance does not result in any hardware changes, nor does it significantly increase the probability of occurrence for initiation of any analyzed events since the function of the equipment has remained unchanged. The proposed frequency has been determined to be adequate based on industry operating data as supported by the conclusions reached in NUREG 1366 and NRC GL [Generic Letter] 93–05.

Surveillance tests are intended to provide assurance of continued component operability. The frequency of performance of a surveillance does not significantly increase the consequences of an accident, as a change in frequency does not change the response of the equipment in performing its specified function (i.e. the overall functional capabilities of the rod control system will not be modified). Increasing the interval of control rod exercise testing will reduce the possibility of inadvertent testing related [to] reactor trips and dropped rods, and resulting in fewer challenges to safety systems and resultant plant transients.

This change does not involve a significant increase in the consequences of an accident or event previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed revision. Existing system and component redundancy and operation is not being changed by the proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated. Therefore, this change does not affect the consequences of previously evaluated accidents.

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

This change does not introduce nor increase the number of failure mechanisms of a new or different type of accident than those previously evaluated since there are no physical changes being made to the facility. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. All 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 nor result in any increase in challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendment 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 involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in the margin of safety because existing component redundancy is not being changed by this proposed change. There are no changes to initial conditions contributing to accident severity or consequences. The proposed surveillance frequency, as supported by past test results, continues to provide the required assurance of operability, such that safety margins established through the design and facility license, including the Technical Specifications, remain unchanged. Therefore, there are no significant reductions in a margin of safety introduced by this proposed amendment.

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: Claudia M. Craig.

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 notice was previously published as a separate individual notice. The notice content was the same as above. It was published as an individual notice either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. It is 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.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: October 20, 1999.

Brief description of amendment: The amendment changed the footnote to the Improved Technical Specifications associated with the Design Features Fuel Storage Specification 4.3.1.1.b which required that 2300 ppm boron be maintained in the Spent Fuel Pool.

Date of publication of individual notice in **Federal Register**: November 19, 1999 (64 FR 63346).

Expiration date of individual notice: December 20, 1999.

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, the Gelman Building, 2120 L

Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

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

Date of application for amendments: September 14, 1999.

Brief description of amendments: The amendments approve the administrative changes to PVNGS TS 5.5.2, Primary Coolant Sources Outside Containment, to delete the references to the post-accident sampling return piping of the radioactive waste gas system and the liquid radwaste system, and TS 5.6.2, Annual Radiological Environmental Operating Report, to delete the administrative requirement to include in the report certain TLD [thermoluminescence dosimeter] results that are no longer available.

Date of issuance: November 24, 1999.

Effective date: November 24, 1999, to be implemented within 60 days.

Amendment Nos.: Unit 1—122, Unit 2—121, Unit 3—121.

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

Date of initial notice in **Federal Register**: October 20, 1999 (64 FR 56528).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 24, 1999.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: October 21, 1999.

Brief description of amendment: This amendment revises Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant by implementing selected improvements described in NRC Generic Letter (GL) 93–05, ''Line-Item Technical Specifications To Reduce Surveillance Requirements For Testing During Power Operation,'' dated September 27, 1993.

Date of issuance: December 17, 1999.

Effective date: December 17, 1999.

Amendment No: 93.

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

Date of initial notice in **Federal Register**: November 17, 1999 (64 FR 62705).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 17, 1999.

No significant hazards consideration comments received: No.

CBS Corporation, Docket No. 50–22, Westinghouse Test Reactor, Waltz Mill, Pennsylvania

Date of application for amendment: September 15, 1999, as supplemented on October 4, 1999.

Brief description of amendment: This amendment changes the decommissioning Technical Specifications dealing with controls for ingress, egress, and equipment removal from containment.

Date of issuance: December 7, 1999.

Effective Date: December 7, 1999.

Amendment No: 11.

Facility License No. TR–2: This amendment changes the decommissioning Technical Specifications.

Date of initial notice in **Federal Register**: November 3, 1999 (64 FR 59798).

The Commission has issued a Safety Evaluation for this amendment dated December 7, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 20, 1999.

Brief description of amendment: The amendment would revise the Technical Specifications to change the senior license requirements for the Operations Manager.

Date of issuance: December 15, 1999.

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

Amendment No: 46.

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

Date of initial notice in **Federal Register:** September 2, 1999 (64 FR 49027).

The July 20, 1999, letter providing clarifying information that did not change the scope of the original application and proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 15, 1999.

No significant hazards consideration comments received: No.

Duquesne Light Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: May 5, 1999, as supplemented June 22 and July 30, 1999.

Brief description of amendments: These amendments conform the licenses to reflect the transfer of Operating Licenses Nos. DPR–66 and NPF–73 for the Beaver Valley Power Station Unit Nos. 1 and 2, to the extent held by Duquesne Light Company (DLC) to the Pennsylvania Power Company, and the operating authority under the licenses from DLC to FirstEnergy Nuclear Operating Company as previously approved by an Order dated September 30, 1999.

Date of issuance: December 3, 1999.

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

Amendment Nos.: 226 and 104.

Facility Operating License Nos. DPR–66 and NPF–73: These amendments revised the Operating Licenses.

Date of initial notice in **Federal Register**: June 14, 1999 (64 FR 31880).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 30, 1999. The June 22 and July 30, 1999, supplements were within the scope of the initial application as originally noticed.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 14, 1999.

Brief description of amendment: This amendment eliminates License Condition 2.C.10 of the Operating License regarding controls over the containment air locks during plant outages and modifies License Condition 2.F of the Operating License regarding reporting requirements for violations of the Technical Specifications and the Environmental Protection Plan.

Date of issuance: December 15, 1999.

Effective date: December 15, 1999.

Amendment No.: 109.

Facility Operating License No. NPF–58: This amendment revised the Operating License.

Date of initial notice in **Federal Register**: November 3, 1999 (64 FR 59803).

The Commission's related evaluation of the amendment is contained in a

Safety Evaluation dated December 15, 1999.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc., et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: June 29, 1999, as supplemented August 27, October 29, and November 3, 1999.

Brief description of amendment: The amendment clarifies the authority to possess certain types of radioactive materials and components at either Unit 1 or Unit 2. Following the transfer of the Three Mile Island, Unit 1 (TMI–1), operating license to AmerGen, these items, under the amendment, may continue to be moved between the TMI–1 and TMI–2 units as they currently are.

Date of issuance: December 9, 1999.

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

Amendment No.: 217.

Facility Operating License No. DPR–50: Amendment revised the License.

Date of initial notice in **Federal Register**: July 12, 1999 (64 FR 37572). The August 27, October 29, and November 3, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 9, 1999.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, (TMI–2) Middletown, Pennsylvania

Date of application for amendment: June 29, 1999, as supplemented by letters dated August 27, October 29, and November 3, 1999.

Brief description of amendment: The amendment adds a provision to the license conditions to ensure that the storage of certain types of radioactive materials and components at Three Mile Island (TMI), Unit 2, pursuant to the TMI, Unit 1 license, does not result in a source term that would exceed the limits in the TMI, Unit 2 Post-Defueling Monitored Storage Safety Analysis Report.

Date of issuance: December 14, 1999.

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

Amendment No.: 53.

Facility Operating License No. DPR–73: Amendment revised the License.

Date of initial notice in **Federal Register**: July 12, 1999 (64 FR 37572). The August 27, October 29, and November 3, 1999, supplements provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 14, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 23, 1999, as supplemented July 30, August 9, August 20, October 7, and October 11, 1999.

Brief description of amendment: The amendment replaces references to Illinois Power Company in the Operating License with references to AmerGen Energy Company, LLC, to reflect the transfer of the license as approved by an Order dated November 24, 1999.

Date of issuance: December 15, 1999.

Effective date: December 15, 1999.

Amendment No.: 123.

Facility Operating License No. NPF–62: The amendment revised the Operating License.

Date of initial notice in **Federal Register**: August 19, 1999 (64 FR 45290).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 24, 1999.

Comments received: Yes. Comments received from The Environmental Law and Policy Center of the Midwest were addressed in the staff's safety evaluation.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: September 23, 1999, as supplemented October 11 and November 10, 1999.

Brief description of amendments: The amendments provide approval to move steam generator sections through the auxiliary building and to disengage crane travel interlocks, and provide relief from performance of Technical Specification Surveillance Requirement 4.9.7.1. The loads to be moved are in support of the Unit 1 Steam Generator Replacement Project.

Date of issuance: December 7, 1999.

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

Amendment Nos.: 233 and 216.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 26, 1999 (64 FR 57665). The October 11, 1999, submittal provided corrected TS pages. The November 10, 1999, submittal was in response to a NRC request for additional information dated October 26, 1999, and provided clarifying information to the original submittal. This information was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: October 1, 1999, as supplemented November 19, 1999.

Brief description of amendments: The amendments involve the resolution of an unreviewed safety question related to certain small-break loss-of-coolant accident scenarios for which there may not be sufficient containment recirculation sump water inventory to support continued operation of the emergency core cooling system and containment spray system pumps during and following switchover to cold leg recirculation. Resolution of this issue consists of a combination of physical plant modifications, new analyses of containment recirculation sump inventory, and resultant changes to the accident analyses to ensure sufficient water inventory in the containment recirculation sump. The amendments would also change the Technical Specifications dealing with the refueling water storage tank inventory and temperature, the required amount of ice in each ice basket in the containment, and the delay to start the containment air recirculation/hydrogen skimmer fans.

Date of issuance: December 13, 1999.

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

Amendment Nos.: 234 and 217.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: October 29, 1999 (64 FR 58458).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 1999.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: February 12, 1999.

Brief description of amendment: The amendment changes the Technical Specifications to (1) allow reactor vessel hydrostatic and leakage tests when reactor coolant temperature is above 212°F without maintaining primary containment integrity and (2) establish a limit and a surveillance requirement on reactor coolant activity when reactor coolant temperature is above 212°F, the reactor is not critical, and primary containment has not been established.

Date of issuance: November 24, 1999.

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

Amendment No.: 107.

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

Date of initial notice in **Federal Register**: March 24, 1999 (64 FR 14283).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 24, 1999.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: September 30, 1999.

Brief description of amendment: The amendment changes the Technical Specification surveillance periodicity requirements for the control room emergency filtration system.

Date of issuance: December 8, 1999.

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

Amendment No.: 108.

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

Date of initial notice in **Federal Register**: November 3, 1999 (64 FR 59805).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 8, 1999.

No significant hazards consideration comments received: No.

PECO Energy Company, Public Service Electric and Gas Company,Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: March 1, 1999, as supplemented June 14, October 1 and October 6, 1999.

Brief description of amendment: The amendment supports the installation of a digital Power Range Neutron Monitoring system and the incorporation of the long-term thermal-hydraulic stability solution hardware.

Date of issuance: October 14, 1999.

Effective date: Effective as of date of issuance and shall be implemented prior to restart from the Peach Bottom Atomic Power Station, Unit 3, October 1999 refueling outage.

Amendment No.: 234.

Facility Operating License No. DPR–56: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: June 2, 1999 (64 FR 29711). The June 14, October 1 and October 6, 1999, 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 October 14, 1999.

No significant hazards consideration comments received: No.

Power Authority of The State of New York, Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: April 6, 1999.

Brief description of amendment: The amendment changes the Technical Specifications by removing the words ''three individual underground'' and ''underground'' from the limiting conditions for operation when referring to the emergency diesel generator fuel oil storage tanks in Sections 3.7.A.5 and 3.7.F.4.

Date of issuance: December 7, 1999.

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

Amendment No.: 198.

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

Date of initial notice in **Federal Register**: June 2, 1999 (64 FR 29713).

No significant hazards consideration comments received: No.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

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

Dates of amendments request: March 12, 1998, as supplemented by letters of April 24, 1998, August 20, 1998, November 20, 1998, February 3, 1999, February 20, 1999, April 30, 1999 (two letters), May 28, 1999, June 30, 1999, July 27, 1999, August 19, 1999, August 30, 1999, September 15, 1999, and September 23, 1999.

Brief description of amendments: The amendments fully convert SNC's Current TS (CTS) to Improved TS (ITS) based on NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants,'' Revision 1, of April 1995. The amendments add two new Additional Conditions to Appendix C of the Unit 1 and Unit 2 Facility Operating Licenses. The first new Additional Condition authorizes SNC to relocate certain CTS requirements to SNC-controlled documents. The second new condition addresses the schedule for performing new and revised ITS surveillances.

Date of issuance: November 30, 1999.

Effective date: As of the date of issuance and shall be implemented no later than March 31, 2000.

Amendment Nos.: 146 and 137.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments fully convert SNC's CTS to ITS.

Dates of initial notices in **Federal Register:** May 25, 1999 (64 FR 28218) and August 25, 1999 (64 FR 46443). The supplemental letters dated April 24, 1998, August 20, 1998, November 20, 1998, February 3, 1999, February 20, 1999, April 30, 1999 (two letters), May 28, 1999, June 30, 1999, July 27, 1999, August 19, 1999, August 30, 1999, September 15, 1999, and September 23, 1999, provided clarifying information that did not change the initial proposed no significant hazards consideration determinations.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 30, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 28, 1999.

Brief description of amendments: The amendments revised Vogtle's operating licenses to allow the licensee to establish containment hydrogen monitoring within 90 minutes of initiation of a safety injection following a loss-of-coolant accident, compared to the current 30 minute requirement.

Date of issuance: December 8, 1999.

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

Amendment Nos.: 110 and 88.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Operating Licenses.

Date of initial notice in **Federal Register:** August 11, 1999 (64 FR 43779).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1999.

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: July 28, 1998, as supplemented by letters dated May 31 and October 21 (2 letters), 1999.

Brief description of amendments: The amendments authorize the revision of the South Texas Project updated final safety analysis report (UFSAR) to allow the use of operator action to reduce the steam generator power-operated relief valve setpoint consistent with the revised small-break loss-of-coolant accident analysis for the replacement Delta 94 SGs.

Date of issuance: December 14, 1999.

Effective date: December 14, 1999. Revisions will be incorporated into the next UFSAR update in accordance with the schedule in 10 CFR 50.71(e).

Amendment Nos.: Unit 1—119, Unit 2—107.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments authorize revision of the UFSAR.

Date of initial notice in **Federal Register:** September 9, 1998 (63 FR 48268).

The May 31 and October 21 (2 letters), 1999, supplements provided additional clarifying information. One of the October 21, 1999, supplements also provided a revised UFSAR pages. This information was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 14, 1999.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant , Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: September 30, 1999.

Description of amendment request: The amendments revise the operating licenses to remove license conditions that have become outdated, are no longer applicable, or are redundant, and to consolidate license conditions which currently exist in two locations in each units license.

Date of issuance: December 16, 1999.

Effective date: December 16, 1999.

Amendment Nos.: 237, 262, and 222.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68: Amendments revised the licenses.

Date of initial notice in **Federal Register:** November 3, 1999 (64 FR 59807).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 1999.

No significant hazards consideration comments received: No.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: February 27, 1998, as supplemented by letters dated June 10, 1998, and October 22, 1999.

Brief description of amendments: The amendments change the refueling water storage tank (RWST) low-low level setpoints in Technical Specification Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' to increase the volume of water available to containment spray pumps when the containment spray system switches to the recirculation mode of operation.

Date of issuance: December 8, 1999.

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

Amendment Nos.: 73 and 73.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 15, 1998 (63 FR 38205). The October 22, 1999, supplement provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application beyond the scope described in the initial notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 8, 1999.

No significant hazards consideration comments received: No.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: August 18, 1999.

Brief description of amendment: The amendment revises the reactor core spiral reloading pattern such that it begins around a source range monitor. The offloading pattern is the reverse sequence.

Date of Issuance: December 14, 1999.

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

Amendment No.: 181.

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

Date of initial notice in **Federal Register:** September 8, 1999 (64 FR 48867).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated December 14, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 23, 1998.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) by deleting the test requirements for snubbers from the TSs. These requirements are already included in the Point Beach Nuclear Plant In-Service Inspection Program.

Date of issuance: December 6, 1999.

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

Amendment Nos.: 191 and 196.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 30, 1998 (63 FR 71977).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 6, 1999.

No significant hazards consideration comments received: No.

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

Date of amendment request: December 29, 1998, as supplemented by letters dated July 29 and October 21, 1999.

Brief description of amendment: The amendment revised (1) the reactor coolant system (RCS) heatup and cooldown limit curves in Figures 3.4–2 and 3.4–3 and cold overpressure mitigation system power-operated relief valve setpoint limit curve in Figure 3.4–4 of the current TSs, and (2) the list of references in Section 5.6.6 on the RCS pressure temperature limits report (PTLR) in the improved TSs. The improved TSs were issued in Amendment No. 123, dated March 31, 1999, to replace the current TSs, but have not yet been implemented. The revision to Section 5.6.6 of the improved TSs replaced the previous references to NRC documents giving criteria for the above limit curves in the current TSs by the references to (1) the NRC letter of December 2, 1999, that approved the use of the PTLR of Generic Letter 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,'' dated January 31, 1996, for WCGS, and (2) WCAP–14040–NP–A, ''Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves.'' The PTLR will provide the methodology for the licensee to revise the heatup and cooldown and setpoint limit curves for WCGS in the future without prior staff approval, after the improved TSs are implemented and have replaced the current TSs. The improved TSs are to be implemented by December 31, 1999.

Date of issuance: December 7, 1999.

Effective date: December 7, 1999, to be implemented by December 31, 1999.

Amendment No.: 130.

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

Date of initial notice in **Federal Register:** February 24, 1999 (64 FR 9023) and September 8, 1999 (64 FR 48869). The October 21, 1999, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 7, 1999.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 8, 1999.

Brief description of amendment: The amendment corrects 15 errors in the improved Technical Specifications that was issued in Amendment No. 123 on March 31, 1999. In addition, four corrections to Table LG, ''Details Relocated from Current Technical Specifications [CTS],'' that was attached to the safety evaluation dated March 31, 1999, issued with Amendment No. 123 were made.

Date of issuance: December 16, 1999.

Effective date: December 16, 1999, to be implemented December 31, 1999.

Amendment No.: 131.

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

Date of initial notice in **Federal Register:** November 16, 1999 (64 FR 62231).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 16, 1999.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of December 1999.

For the Nuclear Regulatory Commission.

Suzanne C. Black,

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

[FR Doc. 99–33684 Filed 12–28–99; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Privacy Act of 1974; Amendment to a System of Records

AGENCY: Office of Personnel Management (OPM).

ACTION: Notice of a new system of records.

SUMMARY: OPM proposes to add a new system of records to its inventory of records systems subject to the Privacy Act of 1974 (5 U.S.C. 552a), as amended. This action is necessary to meet the requirements of the Privacy Act to publish in the **Federal Register** notice of the existence and character of record systems maintained by the agency (5 U.S.C.552a(e)(4)).

DATES: The changes will be effective without further notice February 8, 2000, unless comments are received that would result in a contrary determination.

ADDRESSES: Send written comments to the Office of Personnel Management, ATTN: Mary Beth Smith-Toomey, Office of the Chief Information Officer, 1900 E Street NW., Room 5415, Washington, DC 20415–7900.

FOR FURTHER INFORMATION CONTACT: Mary Beth Smith-Toomey, (202) 606-8358.

SUPPLEMENTARY INFORMATION: The photo identification and visitor access records system was established to improve security in OPM facilities. This system allows the system manager to control and/or monitor access to the building and sensitive areas within the building.

Office of Personnel Management.

Janice R. Lachance,

Director.

OPM/INTERNAL–14

SYSTEM NAME:

Photo Identification and Visitor Access Control Records.

SYSTEM LOCATION:

U.S. Office of Personnel Management, Office of Contracting and Administrative Services, 1900 E Street NW., Washington, DC 20415–7100.

CATEGORIES OF INDIVIDUALS COVERED BY THE SYSTEM:

Individuals visiting OPM facilities, OPM employees, contractors, and retirees seeking access to OPM facilities and classified records.

CATEGORIES OF RECORDS IN THE SYSTEM:

Records of individuals visiting OPM and employees, contractors, and retirees identification files (including photographs) maintained for access purposes.

AUTHORITY FOR MAINTENANCE OF THE SYSTEM:

Federal Property and Administrative Services of 1949, as amended, and 40 U.S.C. 486(c).

ROUTINE USES OF RECORDS MAINTAINED IN THE SYSTEM, INCLUDING CATEGORIES OF USERS AND THE PURPOSES OF SUCH USES:

Routine use 1 of the Prefatory Statement at the beginning of OPM's