STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 24

Tuesday, July 25

3:25 p.m.

Affirmation Session (Public Meeting)

a. Final Rule to Amend 10 CFR Part 70, Domestic Licensing of Special Nuclear Material

b. Final Rule: 10 CFR Part 72—Clarification and Addition of Flexibility

Week of July 31—Tentative

There are no meetings scheduled for the Week of July 31.

Week of August 7—Tentative

There are no meetings scheduled for the Week of August 7.

Week of August 14—Tentative

Tuesday, August 15

9:25 a.m.

Affirmation Session (Public Meeting) (If necessary)

9:30 a.m.

Briefing on NRC International Activities (Public Meeting) (Contact: Ron Hauber, 301–415–2344)

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

Week of August 21—Tentative

Monday, August 21

1:55 p.m.

Affirmation Session (Public Meeting) If necessary)

Week of August 28—Tentative

There are no meetings scheduled for the Week of August 28.

* THE SCHEDULE FOR COMMISSION MEETINGS IS SUBJECT TO CHANGE ON SHORT NOTICE. TO VERIFY THE STATUS OF MEETINGS CALL (RECORDING)—(301) 415–1292. CONTACT PERSON FOR MORE INFORMATION: Bill Hill (301) 415–1661.

* * * * *

ADDITIONAL INFORMATION: By a vote of 5–0 on March 31, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Intragovernmental Issues'' (Closed Ex. 9) be held on March 31, and on less than one week's notice to the public.

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

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 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: July 21, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–19002 Filed 7–24–00; 12:44 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 July 1, 2000, through July 14, 2000. The last biweekly notice was published on July 12, 2000 (65 FR 43040).

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 August 25, 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. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: June 19, 2000 (U–603367).

Description of amendment request: The proposed amendment would allow some emergency diesel generator (EDG) Technical Specification surveillance requirements to be performed during plant operation instead of during plant shutdown as now required. These EDG surveillance tests include load rejection tests and the EDG 24-hour run test.

Basis for proposed no significant hazards consideration determination: The NRC staff has performed an analysis of the issue of no significant hazards consideration which is presented below:

1. No changes will be made to the design or operation of the emergency diesel generators (EDGs) and the plant electrical distribution system will normally be aligned to minimize perturbations from the EDG tests during power operation. Additionally, while some portions of some surveillance tests will result in a decrease in EDG availability during power operation, EDG availability is not significantly decreased. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. No physical changes will be made to the plant. Electrical protective isolation devices will continue to act as before and Technical Specification system operability requirements are not being changed. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. No changes will be made to the design or operation of the emergency diesel generators (EDGs) and the plant electrical distribution system will normally be aligned to minimize perturbations from the EDG tests during power operation. Additionally, while some portions of some surveillance tests will result in a decrease in EDG availability during power operation, EDG availability is not significantly decreased. Therefore, the proposed change does not significantly reduce a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: June 27, 2000.

Description of amendment request: The proposed amendment would revise Section 3.5.1, ''Safety Injection Tanks (SITs),'' of the Palisades Plant Improved Technical Specifications (ITS) as issued by the NRC on November 30, 1999 (Amendment No. 189), for implementation on or before October 31, 2000. Specifically, Condition A, which currently applies to ''One SIT inoperable due to boron concentration not within limits,'' would be expanded to include ''OR One SIT inoperable due to the inability to verify level or pressure.'' Required Action A.1, which currently states ''Restore boron concentration to within limits,'' would be changed to state ''Restore SIT to OPERABLE status.'' The specified Completion Time for the revised Required Action A.1 would remain as 72 hours. Condition B, which applies to ''One SIT inoperable for reasons other than Condition A,'' would be changed to specify a Completion Time of 24 hours (rather than the current 1 hour) to restore the SIT to OPERABLE status. The licensee also forwarded revised pages to the Palisades ITS Bases for these proposed changes. Additional changes proposed in the licensee's application dated June 27, 2000, (which address the Low-Pressure Safety Injection System) are outside the scope of this **Federal Register** (FR) notice and are addressed in a separate FR notice.

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:

[Operation in Accordance with the Proposed Amendment] Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System. The SITs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

SITs were designed to mitigate the consequences of Loss of Coolant Accidents (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change. In addition, in order to fully evaluate the effect of the SIT Allowable Outage Time (AOT) [a.k.a, ''Completion Time'] extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency or large early release frequency. As a result, from a PSA standpoint, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD–994, Combustion Engineering Owners Group ''Joint Applications Report for Safety Injection Tank AOT/STI [surveillance time interval] Extension.''

The changes pertaining to SIT inoperability based solely on instrumentation malfunction do not involve a significant increase in the consequences of an accident as evaluated and endorsed by the Nuclear Regulatory Commission (NRC) in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements.'' This evaluation is applicable to the Palisades Plant.

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

[Operation in Accordance with the Proposed Amendment] Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change does not change the design, configuration, or method of operation of the plant. The proposed configuration (one SIT out of service) is already allowed. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

[Operation in Accordance with the Proposed Amendment] Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. The proposed configuration (one SIT out of service) is already allowed. PSA evaluations were used to evaluate these changes. The results of these analyses show no significant increase in the core damage frequency or large early release frequency. These evaluations are detailed in CE NPSD–994. Therefore, this change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: June 27, 2000.

Description of amendment request: The proposed amendment would revise Section 3.5.2, ''ECCS [Emergency Core Cooling System]—Operating,'' of the Palisades Plant Improved Technical Specifications (ITS) as issued by the NRC on November 30, 1999 (Amendment No. 189), for implementation on or before October 31, 2000. Specifically, the change would extend the Completion Time (a.k.a., allowed outage time or AOT) for a single low-pressure safety injection (LPSI) subsystem from 72 hours to 7 days. The change would apply for operating Modes 1, 2, and 3 with the primary coolant system temperature at or above 325 degrees F. The licensee also forwarded revised pages to the Palisades ITS Bases for the proposed change.

Additional changes proposed in the licensee's application dated June 27, 2000, (which address the safety injection tanks) are outside the scope of this **Federal Register** (FR) notice and are addressed in a separate FR notice.

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:

[Operation in Accordance with the Proposed Amendment] Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The Low Pressure Safety Injection system (LPSI) is part of the Emergency Core Cooling System. LPSI components are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The LPSI system is primarily designed to mitigate the consequences of a large Loss of Coolant Accident (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change. In addition, in order to fully evaluate the effect of the LPSI AOT extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, from a PSA standpoint, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD–995, Combustion Engineering Owners Group ''Joint Applications Report for Low Pressure Safety Injection System AOT Extension.''

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

[Operation in Accordance with the Proposed Amendment] Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes do not change the design, configuration, or method of operation of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, and the setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed changes do not alter assumptions made in the safety analysis and licensing basis. Therefore, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

[Operation in Accordance with the Proposed Amendment] Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes do not affect the limiting conditions for operation or their bases used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD–995. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed changes. Therefore, this change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: June 21, 2000.

Description of amendment request: The proposed amendments would modify the Emergency Feedwater System (EFW) section of the Updated Final Safety Analysis Report.

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

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

No. The EFW System is utilized to mitigate the consequences of an accident. Failure of the EFW System is not a precursor to any accident evaluated in the UFSAR [Updated Final Safety Analysis Report].

The UFSAR change proposes additional exceptions to the ability of the EFW system to mitigate specific events coupled with a single failure. These exceptions are appropriate, because diverse systems (*i.e.*, the SSF [standby shutdown facility] ASW [auxiliary service water] System or EFW System from another unit) are available to mitigate the defined transient/accident and the probability of the defined transient/accident occurring is small.

The proposed UFSAR changes do not involve any adverse impact on containment integrity, radiological release pathways, fuel design, filtration systems, main steam relief valve setpoints, or radwaste systems. In addition, it does not create any new radiological release pathways.

Therefore, it is concluded that the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

No. The EFW System is utilized to mitigate the consequences of an accident. Failure of the EFW System is not a precursor to any accident evaluated in the UFSAR. The proposed UFSAR changes do not physically effect the plant, nor do they increase the risk of a unit trip or reactivity excursion. This proposed change does not introduce any new accident precursors. Therefore, these proposed changes do not create the possibility of any new or different kind of accident.

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

No. A Probabilistic Risk Assessment (PRA) evaluation of the single failures identified in a failure modes and effects analysis performed for the EFW System concluded that there are no single active failures that contribute significantly to core damage frequency.

The UFSAR change proposes additional exceptions to the ability of the EFW system to mitigate specific events coupled with a single failure. These exceptions are appropriate, because the probability of the defined transient/accident occurring is small, and diverse systems (i.e., the SSF ASW System or EFW System from another unit) are available to mitigate the defined transient/accident.

The proposed UFSAR changes do not involve: (1) a physical alteration of the plant; (2) the installation of new or different equipment; or (3) any impact on the fission product barriers or safety limits.

Therefore, it is concluded that the proposed UFSAR changes will not result in a significant decrease 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: June 29, 2000.

Description of amendment request: The current Arkansas Nuclear One, Unit 2 (ANO–2) Technical Specification (TS) 3.6.2.3 states: ''Two independent containment cooling groups shall be OPERABLE with at least one operational cooling unit in each group.'' The proposed change will modify this wording as follows: ''Two independent containment cooling groups shall be OPERABLE with two operational cooling units in each group.'' In addition, the proposed amendment would change the surveillance requirements contained in TS 4.6.2.3.a. At the present time, TS 4.6.2.3.a. would allow a containment cooler group with a minimum service water flow rate of 1250 gpm to be declared operable if one of the two cooling units and associated fan is operable. As a result of this proposed change, the surveillance requirements will be modified to require both cooling units per group to be operable for the containment cooler group to be operable.

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

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The containment cooling units do not have the ability to cause an accident, however, they do serve to mitigate containment accident conditions. The new MSLB [Main Steam Line Break] and LOCA [Loss of Coolant Accident] analyses contain the same assumptions relating to containment heat removal as the original analyses, i.e., at least one containment building cooling unit in conjunction with one train of CSS [containment spray system] is adequate for containment heat removal. During 2R14 [Unit 2, 14th refueling outage] the containment coolers will be modified by adjusting the fan pitch, which will reduce fan flow as well as the post DBA [Design Basis Accident] motor horsepower. To offset this lower containment cooler fan airflow rate, two cooling units per group will be required. The resulting heat removal capacity with two containment cooling units in service at the new blade pitch position is greater than the required heat removal assumed in the LOCA and MSLB analyses.

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

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

Assuming the single failure of a loss of one group of components, the remaining group with two cooling units will continue to be available. The modification to the fan blade pitch will result in a lower air flow rate through each containment cooler. However, the requirement for two units per group to be operable provides adequate heat removal capacity for containment uprate conditions. Therefore, the heat removal capacity assumed in the Containment Uprate analysis remain valid. The previous ability to credit either cooler unit provided additional design margin whereby the required redundancy is still provided by this change.

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

Criterion 3—Does Not Involve a Significant Reduction in the Margin of Safety.

The Containment Cooling System ensures that (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions. The modification planned during 2R14 will result in a lower air flow rate through each cooling unit and a corresponding reduction in heat transfer capability of each cooling unit. However, the safety margin is still maintained by requiring both cooler units to be operable and thus providing adequate heat removal capacity to remain below the containment design pressure.

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

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

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

NRC Section Chief: Robert A. Gramm.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: June 9, 2000.

Description of amendment request: The proposed amendment would reduce the bypass valve (BV) cycling frequency of SR 3.7.7.1 from 31 days to 92 days. This will make the test frequency for the BVs consistent with the testing frequency for the other Main Turbine Valves (e.g. Main Turbine Control, Stop, and Combined Intermediate Valves). The 92-day frequency is also consistent with the typical testing frequency for stroking safety-related valves under TS 5.5.6, In-Service Testing Program.

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

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

The current TS SR 3.7.7.1 requires that the BVs be cycled once every 31 days to demonstrate that the BVs are mechanically OPERABLE (free to move). DAEC in-house operating experience has shown that the BVs have reliable equipment performance in that they consistently pass the valve cycling test at both the 31-day and 92-day frequency. A test frequency of 92 days is sufficient to ensure the reliability of the BVs. The DAEC is analyzed for certain transient events with the assumption that the MTBS is out-of-service (*e.g.* turbine trips, generator load rejects, feedwater flow controller failure at maximum demand). Continued plant operation is allowed in cases of inoperable MTBS provided the more restrictive MCPR limit is applied (LCO 3.7.7). Margin to the MCPR Safety Limit is bounded by the analyzed failure of the MTBS. Should the BV fail a cycling test, the TS required actions would be taken accordingly. Therefore, this proposed amendment will not involve a significant 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.

There are no modifications made to the MTBS (including BVs) or system operations in this proposed TS amendment. The only change is the BV's cycling frequency from 31 days to 92 days. The proposed TS amendment does not alter the OPERABILITY requirements or performance characteristics of the MTBS. The reduced BV cycling frequency reduces the need for reactivity changes and pressure perturbations on the reactor. This proposed amendment will 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 a margin of safety.

The only change by this proposed TS amendment is the frequency of the BV's cycling test from 31 days to 92 days. The OPERABILITY requirement and functional characteristics of the MTBS remain unchanged. DAEC in-house operating experience has demonstrated that a 92-day test frequency provides reasonable assurance that the BVs remain OPERABLE. The BV's response times are used in determining the effect on the MCPR. The surveillance tests that ensure the MTBS meets the system's automatic actuation requirements (SR 3.7.7.2) and response time limits (SR 3.7.7.3) are not affected by this proposed TS amendment and will continue to be performed at the current TS frequency. Therefore, this proposed amendment will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

IES Utilities Inc., Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: June 14, 2000.

Description of amendment request: Alliant Energy Corporation (AEC) plans to merge and consolidate another utility it owns, Interstate Power Company (IPC), with IES Utilities Inc., effective January 1, 2001. The name of the surviving corporation, IES, would be changed to Interstate Power and Light Company (IP&L).

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.

No physical or operational changes to the DAEC will result from changing the corporate name. The DAEC will continue to be operated in the same manner with the same organization. Therefore, the proposed amendment does not involve a significant 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.

No physical or operational changes to the DAEC will result from changing the corporate name. The DAEC will continue to be operated in the same manner with the same organization. Therefore, the proposed amendment will 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 a margin of safety.

No physical or operational changes to the DAEC will result from changing the corporate name. The DAEC will continue to be operated in the same manner with the same organization. Therefore, the proposed amendment will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Al Gutterman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Claudia M. Craig.

Northeast Nuclear Energy Company, et al., Docket No. 50–245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: June 6, 2000.

Description of amendment request: The proposed amendment would revise the Millstone Nuclear Power Station, Unit No. 1 license to modify or remove license conditions and confirmatory orders to reflect the permanently defueled condition of the unit. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The purpose of the proposed changes is to revise the Millstone Unit No. 1 Operating License to only address conditions and requirements that are germane to the permanently shutdown and defueled condition. Since Millstone Unit No. 1 has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to the operation of the plant are no longer appropriate. Elimination of the unnecessary requirements and statements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant conditions. The proposed changes do not affect the only design basis accident that continues to be applicable (i.e., the fuel handling accident). Therefore, the changes do not increase the probability or consequences of any previously evaluated accident.

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

The purpose of the proposed changes is to revise the Millstone Unit No. 1 Operating License to only address conditions and requirements that are germane to the permanently shutdown and defueled condition. Since Millstone Unit No. 1 has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to the operation of the plant are no longer appropriate. Elimination of the unnecessary requirements and statements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant conditions. The proposed changes do not affect storage of spent fuel. Therefore, the proposed changes do not create a different kind of accident from those previously analyzed.

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

The purpose of the proposed changes is to revise the Millstone Unit No. 1 Operating License to only address conditions and requirements that are germane to the permanently shutdown and defueled condition. Since Millstone Unit No. 1 has permanently ceased operation and will be maintained in a defueled condition, many provisions of the license related to the operation of the plant are no longer appropriate. Elimination of the unnecessary requirements and statements allows the plant staff to focus on those requirements which continue to be appropriate to the existing plant conditions. The proposed changes do not affect storage of spent fuel. Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: Michael T. Masnik

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: June 28, 2000.

Description of amendment request: The proposed changes to the Technical Specifications (TSs) are associated with Section 3/4.7.6, ''Control Room Emergency Ventilation System.'' Specifically, TS 3.7.6.1 will be revised to add a footnote that the Control Room boundary can be opened intermittently under administrative control, and add a new Modes 1 through 4 action requirement that will allow 24 hours to restore the Control Room boundary. In addition, various editorial changes associated with action requirement format and letter designations are proposed.

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

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

The action requirements for the Control Room Emergency Ventilation System have been changed to address the impact a loss of boundary integrity has on the associated system. The proposed changes to the action requirements will not cause an accident. Allowing the Control Room boundary to be opened intermittently under administrative controls will have no adverse impact on the consequences of the design basis accidents since the administrative controls will be able to rapidly restore boundary integrity when required. Allowing 24 hours to restore the Control Room boundary in Modes 1 through 4 could result in an increase in the consequences of a design basis accident to the Control Room personnel. However, considering the low probability of a design basis accident occurring during this time, the proposed allowed outage time is reasonable to allow the boundary integrity to be restored before requiring a plant shutdown.

These changes are consistent with Technical Specification 3.6.5.2, ''Containment Systems—Enclosure Building,'' which allows normal entry and egress through associated access openings (Surveillance Requirement 4.6.5.2.1) and 24 hours to restore Enclosure Building integrity, and with generic industry guidance (NUREG–1432, Technical Specification 3.7.11, TSTF–287, Rev. 5).

The proposed changes to address format issues will not result in any technical changes to the current requirements.

The proposed Technical Specification changes will have no adverse effect on plant operation or the operation of accident mitigation equipment, and will not significantly impact the availability of accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the equipment covered by this specification is not an accident initiator and can not cause an accident. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification changes do not impact any system or component which could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following an accident will not be significantly different as a result of these changes. In addition, the accident mitigation equipment affected by the proposed changes is not an accident initiator. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes to Technical Specification 3.7.6.1 are consistent with Technical Specification 3.6.5.2 which allows normal entry and egress through associated access openings (SR 4.6.5.2.1) and 24 hours to restore Enclosure Building integrity, and with generic industry guidance (NUREG–1432, Technical Specification 3.7.11, TSTF–287, Rev. 5). If the Control Room boundary is not operable, the proposed action requirements will require timely restoration of the boundary or the plant will be placed in a configuration where there is no adverse impact associated with the loss of Control Room boundary integrity. The proposed allowed outage time provides a reasonable time for repairs before requiring a plant shutdown, and reflects the low probability of an event occurring while the boundary is inoperable. The proposed shutdown times, which are consistent with times already contained in the Millstone Unit No. 2 Technical Specifications and with generic industry guidance (NUREG–1432), will allow an orderly shutdown to be performed.

The proposed changes to address format issues will not result in any technical changes to the current requirements. These proposed changes will not adversely impact any of the design basis accidents or the associated accident mitigation equipment.

The proposed changes will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analyses. Therefore, there will be no 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: June 26, 2000.

Description of amendment request: The proposed changes to the Technical Specifications (TSs) are associated with the Reactivity Control Systems section. Specifically, the surveillance requirements associated with the frequency for determining the operability of each rod not fully inserted in the core will be revised from once every 31 days to once every 92 days for Units 2 and 3. In addition, the surveillance requirement associated with the frequency of testing the Control Element Assembly Deviation Circuit will be revised from once every 31 days to once every 92 days for Unit 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to Millstone Unit Nos. 2 and 3 Specification 4.1.3.1.2 will revise the frequency for determining the operability of each rod that is not fully inserted in the core from once every 31 days to once every 92 days. The proposed change in the frequency does not change any of the assumptions used in the safety analyses. On the other hand, the decrease in surveillance frequency will reduce the potential for reactor trips and the unnecessary challenges to the safety systems associated with the performance of the surveillance. Additionally, NNECO [Northeast Nuclear Energy Company] performed Millstone Unit Nos. 2 and 3 specific evaluations of the effect of changing the frequency of rod movement test from 31 days to 92 days on Core Damage Frequency (CDF). These evaluations concluded that the change in test frequency from 31 days to 92 days has no adverse impact on CDF (the estimated potential risk associated with tripping the reactor as a result of this high risk surveillance is about 1.31E–8/yr for Millstone Unit No. 2 and 4.28E–8/yr for Millstone Unit No. 3) and is therefore acceptable. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change in the frequency of testing the CEA Deviation Circuit in Millstone Unit No. 2 Specification 4.1.3.1.3 from once every 31 days to once every 92 days does not change any of the assumptions used in the safety analysis. On the other hand, the decrease in surveillance frequency will reduce the reactor trips and the unnecessary challenges to the safety systems associated with the performance of the surveillance. Additionally, the Deviation Circuit has excellent testing history and increasing the surveillance interval from 31 days to 92 days will have no adverse effect on its overall reliability. The Nuclear Regulatory Commission approved this increase in surveillance interval as part of TSTF [Technical Specifications Task Force] -127.[ ] Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes will not alter [the] configuration of the plants (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plants are operated. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes in the surveillance frequency do not change any of the assumptions used in the safety analyses. Additionally, NNECO performed Millstone Unit Nos. 2 and 3 specific evaluations of the effect of changing the frequency of rod movement test from 31 days to 92 days on CDF. These evaluations concluded that the change in test frequency from 31 days to 92 days has no adverse impact on CDF and is therefore acceptable. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

As described above, this License Amendment Request does not involve a significant increase in the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve an SHC [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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Section Chief: James W. Clifford.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: June 14, 2000.

Description of amendment request: The licensee requests that the Peach Bottom Atomic Power Station (PBAPS), Unit 2, Technical Specifications (TS) contained in Appendix A to the Operating License be amended to: (1) Revise TS 2.1.1.2 to reflect changes in the Safety Limit Minimum Critical Power Ratios (SLMCPRs) due to the cycle-specific analysis performed by Global Nuclear Fuel (formerly General Electric Nuclear Energy (GENE)) for PBAPS, Unit 2, Cycle 14, which includes the use of the GE–14 product line, (2) delete the cycle-specific footnote for the SLMCPRs contained in TS 2.1.1.2 (''Reactor Core SLs''), and (3) update a reference contained in TS 5.6.5.b.2 (''Core Operating Limits Report'') which documents an analytical method used to determine the core operating limits. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The derivation of the cycle specific SLMCPRs for incorporation into the TS, and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999 safety evaluation report.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling. The GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U. S. Supplement, NEDE–24011–P–A–13–US, August, 1996 (GESTAR–II), which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of these changes. Additionally, as a result of the use of the GE–14 product line, no dose calculations are being adversely impacted. Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

In addition to the change to the SLMCPRs, the footnote to TS 2.1.1.2 is being deleted. The footnote associated with TS 2.1.1.2 was originally included to ensure that the SLMCPR value was only applicable for the identified cycle. Since that time, Amendment 25 has been approved. Therefore, this footnote is no longer necessary. The footnote was for information only, and has no impact on the design or operation of the plant. A similar change was previously approved for PBAPS, Unit 3, as discussed in the NRC safety evaluation (Amendment No. 233), dated October 5, 1999. The deletion of the footnote associated with TS 2.1.1.2 is an administrative change that does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The reference to the Revision 1 ARTS/MELLLA analysis contained in TS 5.6.5.b.2 is being updated to a Revision 2 analysis, to reflect changes that were previously approved by the NRC as documented in the safety evaluation report dated August 10, 1994 (Amendment No. 192 for PBAPS, Unit 2). This is an administrative change which will ensure that the references contained in the PBAPS, Unit 2 Technical Specifications are accurate and consistent with other licensing documents. No technical changes are occurring which have not been previously approved by the NRC. A similar change was previously approved for PBAPS, Unit 3, as discussed in the NRC safety evaluation (Amendment No. 233), dated October 5, 1999. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLMCPRs are calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. Additionally, the GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and U.S. Supplement, NEDE–24011–P–A–13–US, August, 1996 (GESTAR–II), which provides the fuel licensing acceptance criteria. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Additionally, this proposed change will delete footnotes contained in TS 2.1.1.2 as the result of the NRC approval of analysis associated with Amendment 25. The proposed change also updates the ARTS/MELLLA analysis reference contained in TS 5.6.5.b.2. This revision contains information which was previously approved by the NRC. Similar changes were previously approved for PBAPS, Unit 3, as discussed in the NRC safety evaluation (Amendment No. 233), dated October 5, 1999. Therefore, the deletion of the footnote associated with TS 2.1.1.2, and the updating of the reference contained in TS 5.6.5.b.2 are administrative changes that do not create the possibility of a new or different kind of accident from any previously evaluated.

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

There is no significant reduction in the margin of safety previously approved by the NRC as a result of: (1) The proposed changes to the SLMCPRs, which includes the use of GE–14 fuel, (2) the proposed change that will delete the footnote to TS 2.1.1.2, and (3) updating the ARTS/MELLLA analysis reference contained in TS 5.6.5.b.2. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–13–US, August 1996, and Amendment 25. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes will not involve a significant reduction in the margin of safety previously approved by the NRC.

Additionally, the proposed changes that delete the footnote to TS 2.1.1.2, and update the revision to the ARTS/MELLLA analysis reference contained in TS 5.6.5.b.2, are administrative changes that will not significantly reduce the margin of safety previously approved by the NRC.

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: J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101

NRC Section Chief: James W. Clifford

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 amendment request: February 4, 2000.

Description of amendment request: The proposed amendment to the Indian Point 3 Technical Specifications (TSs) proposes to revise the main steam line break (MSLB) analysis to correct the assumption for non-isolable feedwater and also to revise assumptions regarding boron in the safety injection piping and assumptions regarding shutdown margin.

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:

Operation of the Indian Point 3 in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92 since it would not:

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

The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated containment pressure for a postulated MSLB. The changes take credit for existing boron in the SI [Safety Injection] system and existing shutdown margin, perform surveillance to verify the boron concentration, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes cannot increase the probability of an accident previously evaluated since they do not change plant operations and are not related to accident initiators. These changes will not increase the consequences of an accident previously evaluated since they do not change system operation to mitigate any accident and the use of a minimum containment test pressure ensures the TS required testing bounds the calculated peak calculated [sic] pressure.

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

The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated pressure. The changes take credit for existing boron in the SI system and existing shutdown margin, perform surveillance to verify the boron, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes do not physically alter the plant since they take credit for existing plant conditions and the physical act of sampling meets system design and Technical Specification requirements. Therefore, these changes do not create the possibility of a new or different type of accident from those previously evaluated.

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

The proposed changes include revised assumptions in the TS to correct non-conservative TS and revised TS with respect to the peak calculated pressure. The changes take credit for existing boron in the SI system and existing shutdown margin, perform surveillance to verify the boron, and revise the containment testing program to reflect a minimum test pressure that must bound the peak calculated pressure. These changes do not involve a significant reduction in the margin of safety since the credited boron is part of the existing system design that has not been credited since the BIT [Boron Injection Tank] tank retirement. The credited shutdown margin is typical of the excess shutdown margin resulting from cycle specific core design.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards considerations. Attorney for licensee: Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

NRC Section Chief: Marsha Gamberoni, Acting.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Units Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: March 13, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Table 3.3–6, ''Radiation Monitoring Instrumentation,'' to change the Containment Gaseous Activity Monitor (R12A) alarm/trip setpoint for the Containment Purge and Pressure Relief system isolation for Mode 6 (Refueling) operation. Specifically, the existing setpoint of less than or equal to two times background would be changed to ''Set at less than or equal to 50% of the 10 CFR [Part] 20 concentration limits for gaseous effluents released to unrestricted areas.'' The proposed amendment will also specify an upper setpoint limit that is not presently required by the existing TS requirement. In addition, the associated TS Bases section would be revised to address the proposed change.

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

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

During Mode 6 operation, the Containment Gaseous Activity Monitor R12A serves to monitor the gaseous activity concentration in the containment atmosphere, and provides an alarm and isolation of the Containment Purge and Pressure Relief system in response to high gaseous activity that would result from a Fuel Handling Accident inside containment. As such, the Containment Gaseous Activity Monitor is not considered as an initiator of any accident previously evaluated. Therefore, the proposed change would not affect the probability of an accident previously evaluated.

The proposed setpoint would allow an alarm/trip setpoint to be higher than the current TS requirements. As a result, it would be expected that the consequences of an accident previously evaluated could possibly increase. However, the proposed setpoint value would isolate the Containment Purge and Pressure Relief system prior to reaching the 10 CFR Part 20 concentration limits for gaseous effluents released to unrestricted areas. The 10 CFR Part 20 limits are equivalent to the radio-nuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce at total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts). These restrictions are intended to minimize and limit the amount of dose received by individual members of the public during normal operations, and are considerably more restrictive than the 10 CFR Part 100 limits. The proposed change would not be considered a significant increase in the consequences of an accident previously evaluated because the revised setpoint would isolate the appropriate release path and maintain doses well below 10 CFR Part 100 limits. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the Containment Gaseous Activity Monitor alarm/trip setpoint will not create any new accident causal mechanisms. Plant operation will not be affected by the proposed amendments and no new failure modes will be created. Thus, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

An evaluation of a fuel handling accident inside containment has been performed by the licensee that demonstrates that the limits of 10 CFR Part 100 would not be exceeded even though no containment isolation was assumed. The analysis assumed that all airborne activity reaching the containment atmosphere would exhaust to the environment within two hours (no containment isolation) and concluded that the exclusion area boundary doses were well within the limits of 10 CFR Part 100. The analysis also demonstrated that the control room doses following the fuel handling accident inside containment would be within General Design Criterion 19 limits. Therefore, the changes proposed by the licensee will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Units Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: May 31, 2000.

Description of amendment request: The proposed amendment would establish new charcoal filter testing requirements for the Auxiliary Building Ventilation (ABV) System, the Control Room Envelope Air Conditioning System (CREACS), and the Fuel Handling Ventilation (FHV) System consistent with the requirements delineated in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999. Specifically, the surveillance requirements associated with Limiting Conditions for Operation (LCOs) 3.7.6.1, 3.7.7.1, and 3.9.12 would specify American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' as the testing methodology.

The May 31, 2000, amendment request would replace Public Service Electric and Gas (PSE&G) Company's original November 24, 1999, application to change Salem Units 1 and 2 Technical Specifications (TS) surveillance requirements associated with the laboratory testing of charcoal samples for the ABV, CREACS, and FHV systems. Additional information associated with the November 24, 1999, submittal was provided on February 10, 2000. However, PSE&G has requested that the November 24, 1999, application be withdrawn.

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 FHV, CREACS and ABV systems will continue to function as designed. The FHV, CREACS and ABV systems are designed to mitigate the consequences of an accident, and therefore, cannot 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 Salem's design basis assumptions are appropriately 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 FHV, CREACS and ABV systems are operated is not affected by these proposed changes. The proposed surveillance requirement acceptance criteria ensure that the FHV, CREACS and ABV safety functions will be accomplished. Therefore, the proposed TS changes would not result in a significant 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 FHV, CREACS and ABV systems are not changed from that currently described in Salem's licensing basis. The FHV, CREACS and ABV 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 FHV, CREACS and ABV systems than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in Salem'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 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 appropriately conservative acceptance criteria. The charcoal filter efficiencies specified in the proposed surveillance requirements apply a safety factor of 2 to the efficiencies used in the design basis dose analysis. There are no increases to the currently approved offsite dose releases or the control room operator doses as a result of these surveillance requirement changes. Therefore, the proposed TS change will not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Units Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: June 14, 2000.

Description of amendment request: The proposed license amendment would allow the use of the Best Estimate Analyzer For Core Operations—Nuclear (BEACON) system at Salem to perform core power distribution measurements. BEACON is a core power distribution monitoring and support system based on a three dimensional nodal code. The system is used to provide data reduction for incore neutron flux maps, core parameter analysis and follow, and core prediction. The licensee has stated that BEACON will be used at Salem to augment the functionality of the flux mapping system when thermal power is greater than 25% of rated thermal power for the purpose of performing power distribution surveillance testing.

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

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

The proposed changes provide a different method for measuring the core power distribution parameters and relocate[s] manufacturing and measurement uncertainty values from the TS [Technical Specifications] to the core operating limits report (COLR). The [TS] power distribution limits themselves are not changed and will continue to be measured and verified to be within limits as required by the current TS surveillances. The cycle-specific core operating limits, although not in TS, will be followed in the operation of the Salem Generating Station. The proposed amendment continues to require the same actions to be taken when or if limits are exceeded as are required by current TS.

Each accident analysis addressed in the Salem Updated Final Safety Analysis Report (UFSAR) will be examined with respect to changes in cycle-dependent parameters, which are obtained from application of the NRC [Nuclear Regulatory Commission]-approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed per requirements of 10 CFR 50.59, ensures that future reloads will not involve an increase in the probability or consequences of an accident previously evaluated.

The method of measuring core power distribution parameters and the location of manufacturing and measurement uncertainty values are not initiators of any previously evaluated accidents and has no influence or impact on the consequences those accidents. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No safety-related equipment, safety function, or plant operation will be altered as a result of the proposed changes. The cycle specific variables are calculated using the NRC-approved methods and submitted to the NRC to allow the Staff to continue to trend the values of these limits. The TS will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded. The change will not introduce any new accident initiators. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes provide a different method for measuring the core power distribution parameters and relocates manufacturing and measurement uncertainty values from the TS to the COLR. The proposed method for measuring the core power distribution parameters has been verified by Westinghouse and reviewed and approved by the NRC. Appropriate measures exist to control the values of the manufacturing and measurement uncertainties. The proposed amendment continues to require operation within the core limits, as obtained from NRC-approved reload design methodologies. Appropriate actions that [are] required to be taken when or if limits are violated remain unchanged. Future changes to measurement and manufacturing uncertainties located in the current TS will be evaluated in accordance with the requirements of 10 CFR 50.59. Since the 10 CFR 50.59 process does not allow any reduction in the margin of safety, prior NRC approval is required prior to a reduction in the margin of safety. Additionally, the Salem TS require revisions of the plant COLR be submitted to the NRC upon issuance. Therefore, the 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.

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

Date of amendment request: June 12, 2000.

Description of amendment request: This amendment would revise the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) to incorporate new temperature and level limits for the ultimate heat sink (UHS) during plant operation in Modes 1–4. These limits are contained in TS Section 3/4.7.5. The minimum required service water pond (SWP) level would be increased from the 415′ elevation to 416.5′ and the maximum allowed temperature at the discharge of the service water pumps would be decreased from 95°F to 90.5°F.

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?

Implementation of the new temperature and level limits for the service water pond do not contribute to the initiation of any accident evaluated in the FSAR [Final Safety Analysis Report]. Supporting factors are as follows:

- The new limits maintain the Service Water System (SWS) design temperature of 95°F during a normal shutdown and post accident and have been developed in accordance with the general requirements of Regulatory Guide 1.27, Revision 2.
- Overall plant performance and operation is not altered by the proposed changes.
- Fluid and auxiliary systems, which are important to safety, are not adversely impacted and will continue to perform their design function.

Therefore, since the reactor coolant pressure boundary integrity and system functions are not impacted, the probability of occurrence of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

The SWP level and temperature limits relate to the plant's ability to reject heat to the ultimate heat sink during normal operation, a normal plant shutdown and hypothetical accident conditions. The new limits preserve the SWS design temperature of 95°F, even during worst case post accident conditions, thus assuring that equipment within the SWS and its interfacing systems remain qualified and that the heat transport capability of the SWS and its interfacing systems [remain] within design values. Since the SWS and its interfacing systems will continue to perform their design functions, it is concluded that the consequences of an accident previously evaluated in the FSAR are not increased.

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

The proposed changes revise the UHS temperature and level limits within TS 3/4.7.5 to incorporate the results of a new thermal analysis performed in accordance with the requirements of Regulatory Guide 1.27, Revision 2. The new limits ensure that SW temperature, as measured at the discharge of the SW pump, [remains] less than the design value of 95°F. No new accident initiator mechanisms are introduced as:

- Structural integrity of the RCS [reactor coolant system] is not challenged.
- No new failure modes or limiting single failures are created.
- Design requirements on all affected systems are met.

Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

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

The proposed changes revise the UHS temperature and level limits [within] TS 3/4.7.5 to incorporate the results of a new thermal analysis performed in accordance with the requirements of Regulatory Guide 1.27, Revision 2. The new limits ensure that SW temperature, as measured at the discharge of the SW pump, [remains] less than the design value of 95°F under both normal and post-accident conditions using the worst case combination of meteorology and operational parameters. Design margins associated with systems, structures and components that are cooled by the SWS are not affected. Since the SWS design temperature is maintained during both normal and worst case accident conditions, the results and conclusions for all design basis accidents remain applicable.

The proposed changes impose more restrictive operating limitations, and their use provides increased assurance that the SWS design temperature will not be exceeded. Since the UHS will continue to provide a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature, it is concluded that the changes do not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: October 13, 1999, as supplemented by letter dated June 1, 2000.

Description of amendment request: The proposed amendments would revise Vogtle's Technical Specification to permit relaxation of allowed bypass test time and completion times for Limiting Conditions for Operations (LCO) 3.3.1, Reactor Trip System Instrumentation and LCO 3.3.2, Engineered Safety Feature Actuation System Instrumentations. These changes specifically revise the completions times from 6 hours to 72 hours for inoperable analog instruments, increase bypass times from 4 hours to 12 hours for surveillance testing of analog channels, and increase completion times from 6 hours to 24 hours for an inoperable logic cabinet or master and slave relays.

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 reactor trip and engineered safety features functions are not initiators of any design basis accident or event, and therefore the proposed changes do not increase the probability of any accident previously evaluated. The proposed changes to the allowed Completion Times and bypass test times do not change the response of the plant to any accidents and have an insignificant impact on the reliability of the reactor trip system and engineered safety feature actuation system (RTS and ESFAS) signals. The RTS and ESFAS will remain highly reliable and the proposed changes will not result in a significant increase in the risk of plant operation. This is demonstrated by showing that the impact on plant safety as measured by core damage frequency (CDF) is less than 1.0E–06 per year and the impact on large early release frequency (LERF) is less than 1.0E–07 per year. In addition, the incremental conditional core damage probabilities (ICCDP) and incremental conditional large early release probabilities (ICLERP) are less than 5.0E–08. These increases/values meet the acceptance criteria in Regulatory Guide 1.174 and 1.177. Therefore, since the RTS and ESFAS will continue to perform their functions with high reliability as originally assumed, and the increase in risk as measured by CDF, LERF, ICCDP, ICLERP is within the acceptance criteria of existing regulatory guidance, there will not be a significant increase in the consequences of any accidents.

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

The proposed changes do not result in a change in the manner in which the RTS and ESFAS provide plant protection. The RTS and ESFAS will continue to have the same setpoints after the proposed changes are implemented. There are no design changes associated with the license amendment. The changes to Completion Times or increased bypass test times do not change any existing accident scenarios nor create any new or different accident scenarios. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. Safety analysis acceptance criteria are not impacted. Redundant RTS and ESFAS trains are maintained, and diversity with regard to signals to provide reactor trip and engineered safety features actuation will be maintained. All signals credited as primary or secondary, and all operator action credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria in Regulatory Guide 1.174 and 1.177. Although there was no attempt to quantify any positive human factors benefit due to increased Completion Times and bypass test times, it is expected that there would be a net benefit due to a reduced potential for spurious reactor trips and actuations associated with testing. Therefore, the proposed license amendment does not involve a significant reduction in margin of safety.

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

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

Acting Section Chief: L. Raghavan, Acting.

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: September 28, 1998, as revised on April 22, 1999, and April 27, 2000. This application was originally noticed on November 18, 1998 (63 FR 64122).

Description of amendment request: The proposed amendments would modify the requirements associated with the control room and fuel handling building heating, ventilation, and air conditioning systems by adding an allowed outage time of 12 hours for a condition where multiple trains of the control room and fuel handling building heating, ventilation, and air conditioning systems are inoperable. The proposed amendments also include changes to make the required action for the affected ventilation actuation instrumentation consistent with the action for inoperable ventilation trains. In addition, the proposed amendments include minor administrative changes to remove an expired dated action and to provide consistency of terminology used in the Technical Specifications (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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes do not involve an [significant] increase in the probability or consequences of an accident previously evaluated. The proposed changes consist of:

(a) Assuring that the Specifications define consistent allowed outage times when the same safety function is addressed in multiple Specifications,

(b) Allowing a system to remain inoperable when appropriately restrictive administrative controls are placed on operations that could result in a challenge to the safety function of the system,

(c) Providing an appropriately short Allowed Outage Time for inoperability needed to permit required maintenance and testing that affects all trains of a system,

(d) Redefining system operability and associated actions in a manner consistent with the system design and function,

(e) Aligning a system to the actuated condition on the loss of an actuation channel,

(f) Using consistent terminology throughout the Specifications.

The proposed changes do not represent significant increases in the probability or consequences of an accident because:

(a) The alignment of the action times between actuating system and actuated system operability requirements do not affect probability or consequences since inoperability of the actuated system has the same effect as inoperability of the actuating system. Since the changes proposed to the actuating system action times will reflect those of the actuated system action times, no change to the allowed outage time applicable to the safety function addressed and fulfilled by both, will occur.

(b) Administrative controls to prevent the conduct of operations that could lead to a challenge to the safety function of the system when the actuation system is inoperable, assures that the design bases functions of the system will not be challenged. Therefore, the probability or consequences of an event previously identified have not been significantly changed.

(c) Allowing up to 12 hours to recover from the inoperability of all 3 trains of Control Room Envelope HVAC [heating, ventilation, and air conditioning] or 2 or more trains of Fuel Handling Building HVAC does not represent a significant change to the probability of an accident. The inoperability of the Fuel Handling Building HVAC systems is not identified as a precursor to a design basis event. The inoperability of the Control Room Envelope HVAC is not a percursor to any event previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. With respect to the PRA [probabilistic risk assessment] analysis for Control Room Envelope HVAC, the allowed outage time provides sufficient time to restore Control Room Envelope HVAC to the rooms serving the Reactor Protection System before any detrimental effects would occur or to place the plant in MODE 3 if Control Room Envelope HVAC could not be restored. The low likelihood of a design basis accident during the limited period of allowed inoperability of these systems does not involve a significant increase in the consequences of an accident. The proposed required actions to suspend all operations involving movement of spent fuel, and crane operations with loads over the spent fuel pool reduce the potential for accident initiation during the allowed outage time.

(d) The redefinition of plant operability requirements into functional trains rather than individual components does not affect the required system functional operability. Therefore, this change does not involve an increase in the probability or consequences of an accident previously identified.

(e) The alignment of the Control Room Envelope HVAC System to the same configuration it would be placed in from an actuation of the inoperable radiation monitoring channel places the system in the design condition. This alignment would result in maintaining the control room envelope pressurized and increases the protection afforded to the operators.

(f) The change in terminology does not change any requirements or actions in the Specification. Therefore this change does not represent an increase in the probability or consequences of any accident previously evaluated.

(g) Revising the applicability of Technical Specification ACTION b. in MODES 5 and 6 will add clarity to the specification and make it better reflect STP's three train design. The clarification provides some additional assurance that the system will perform as assumed in the analyses.

Based on the above discussion, the individual changes do not represent an [significant] increase in the probability or consequences of any accident previously evaluated.

In addition to the changes proposed to controls over Control Room Envelope HVAC, Fuel Handling Building HVAC, and associated actuation logic, an administrative change is proposed to remove the footnotes at the bottom of pages 3/4 3–28, 3/4 7–19, and 3/4 7–20. Since the footnotes no longer have meaning or relevance to the operation of the facility, their removal does not increase 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 changes make the existing Specifications internally consistent, manually align a system to the actuated position, provide an alternative measure that assures [that] a safety function which is unavailable is not required to [be] perform[ed], provide an extended period of allowance for all trains of a system to be inoperable, and redefines system operability to reflect its functional design. The proposed changes do not introduce any new equipment into the plant or significantly alter the manner in which existing equipment will be operated. The limited allowed outage time of three inoperable Control Room Envelope HVAC systems has no detrimental effect on the operation of the Reactor Protection System. The systems affected by the proposed changes are not identified as contributing causal factors in design basis accidents; their function is to assist in mitigation of accidents postulated to occur. Since the proposed changes do not allow activities that are significantly different from those presently allowed, no possibility exists for a new or different kind of accident from those previously evaluated.

In addition to the changes proposed to controls over Control Room Envelope HVAC, Fuel Handling Building HVAC, and associated actuation logic, an administrative change is proposed to remove the footnotes at the bottom of pages 3/4 3–28, 3/4 7–19, and 3/4 7–20. Since the footnotes no longer have meaning or relevance to the operation of the facility, their removal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety because the ability of the Fuel Handling Building HVAC and Control Room Envelope HVAC Systems to perform their function will be maintained. The margin of safety is defined by the ability of the systems to limit the release of radioactive materials and limit exposures to operators following a postulated design basis accident. The only aspect of the proposed change that can be postulated to have any effect on a margin of safety is the proposed allowance for all trains of Control Room Envelope HVAC or Fuel Handling Building HVAC to be inoperable for a limited period. The low probability of a design basis event that would require the system to perform its safety function during the limited period allowed by the proposed action assures that the change does not involve a significant change in a margin of safety. Therefore, the proposed changes do not significantly affect these operating restrictions and the margin of safety which support the ability to make and maintain the reactor in a safe shutdown and limit the release of radioactive material is not affected.

Sufficient time is allowed to restore Control Room Envelope HVAC to the rooms serving the Reactor Protection System before any detrimental effects would occur or to place the plant in MODE 3 if Control Room Envelope HVAC could not be restored.

Revising the applicability of Technical Specification 3.7.7 ACTION b. in MODES 5 and 6 will add clarity to the specification, make it better reflect STP's three train design and provide greater assurance that desired margins are maintained.

Suspending fuel movement and crane operations with loads over the spent fuel pool when all Fuel Handling Building or Control Room Envelope HVAC systems are inoperable prevents a Fuel Handling Accident from occurring, which maintains the margin of safety for this design event.

In addition to the changes described above, an administrative change is proposed to remove the footnotes at the bottom of pages 3/4 3–28, 3/4 7–19, and 3/4 7–20. Since these footnotes are no longer applicable to the facility, their removal cannot result in 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 standards of 10 CFR 50.92(c) are satisfied. The staff also reviewed the proposed change to provide consistency of terminology in the TSs for no significant hazards consideration. This proposed administrative change does not affect the design or operation of the facility and satisfies the three standards of 10 CFR 50.92(c). 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 amendment request: June 22, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to remove the applicability of core alteration requirements from those TS that are designed to mitigate the consequences of a fuel handling accident. The applicable TS bases would also be revised.

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

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

The proposed revision eliminates requirements associated with core alterations for specifications that are intended to mitigate the consequences of a fuel handling accident (FHA). These functions will not impact accident generation because their function is to support mitigation of accidents and they are not considered to be the source of a postulated accident. The removal of these actions and surveillance requirements affects functions that are not necessary during core alterations because postulated events during these activities do not have the potential to result in major fuel cladding damage like that assumed for an FHA. Therefore, there is no adverse impact to nuclear safety by eliminating core alteration requirements for specifications that provide for the mitigation of an FHA.

The proposed revision also clarifies the use of equivalent methods for isolation of containment penetrations. Equivalent isolation methods will maintain acceptable isolation capability for postulated conditions that could occur during the movement of irradiated fuel. This change does not alter the current intent or expectations for containment closure requirements during the movement of irradiated fuel and only serves to delineate other methods that provide an acceptable level of isolation. The status of penetration isolation methods during fuel movement does not impact the generation of an accident. This is based on these functions only providing a radiation barrier in the event of an FHA and not as a potential initiator for postulated accidents.

Based on the previous discussions, the proposed revision does not alter any plant equipment or operating practices; therefore, the probability of an accident is not significantly increased. In addition, the consequences of an accident is not significantly increased by eliminating core alteration requirements for specifications that only support the mitigation of FHAs or by using equivalent isolation methods for containment penetrations. This is based on sufficient safety function capabilities being available for the mitigation of an FHA or other potential events that could occur during core alteration activities.

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

The proposed allowance to eliminate core alteration requirements for FHA related specifications and utilize equivalent isolation methods for containment penetrations will not alter plant functions or equipment operating practices. The proposed elimination of core alteration requirements will not impact accident generation because these functions provide for FHA mitigation and are not postulated to be an initiator of postulated accidents. Containment penetration isolation methods are not considered to be the source of a postulated accident. Therefore, since plant functions and equipment are not altered and the availability of FHA mitigation functions and isolation of containment penetrations do not contribute to the initiation of postulated accidents, the proposed revision will not create a new or different kind of accident.

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

The elimination of core alteration requirements for specifications that provide mitigation functions for FHAs will not affect the ability of these functions to perform as necessary. This is based on postulated events during core alteration not having the potential to result in fuel cladding damage that is assumed for the FHA and therefore, not requiring functions necessary to mitigate the FHA event. The proposed revision will continue to provide acceptable provisions for activities that could result in an FHA or events postulated during core alterations to maintain the necessary margin of safety.

The equivalent methods for containment penetration isolation provide the same level of isolation for conditions that may occur during fuel movement. Therefore, the equivalent isolation methods provide an acceptable barrier to the release of radiation as do the other listed methods and maintains the required margin of safety.

Therefore, the margin of safety provided by the containment building penetration requirements and other specifications for the mitigation of FHAs is not significantly reduced by the proposed allowance to eliminate affected core alteration requirements or to use equivalent methods for containment penetration isolation.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H Knoxville, Tennessee 37902 .

NRC Section Chief: Richard P. Correia.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: June 22, 2000.

Description of amendment request: The proposed changes will modify the Technical Specifications in Sections 3.1.2.7, 3.1.2.8, 3.5.1, 3.5.5, 3.6.2.2, 3.9.1, and associated Bases Sections to allow for an increase of boron in the refueling water storage tank (RWST), casing cooling tank (CCT), spent fuel pool, and safety injection accumulators (SIAs).

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.

Increased boron concentration limits for the RWST, CCT, SIAs, and Spent Fuel Pool (SFP) will not increase the consequences of an accident previously evaluated. The increased boron concentration limits reduce the time to switchover from cold to hot leg recirculation, which will prevent boron precipitation in the reactor vessel following a loss of coolant accident (LOCA). The post-LOCA sump boron concentration limit is revised to ensure adequate post-LOCA shutdown margin. The post-LOCA containment sump and quench spray (QS) pH remain within the limits specified in the Standard Review Plan. All other transients either were not impacted or were made less severe as a result of the increased boron concentrations.

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

The proposed increase in boron concentration does not add new or different equipment to the facility, nor does this change the manner that plant equipment is being operated. Although the increased boron concentration requires procedure changes to ensure that cold to hot leg (reactor coolant loops) recirculating after an accident occurs earlier in the event, there are no changes to the methods utilized to respond to plant transients. The proposed Technical Specification changes do not alter instrumentation setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. Therefore, the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report is not created.

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

The LOCA considerations, including the recirculation switchover time, the post-LOCA sump boron concentration limit, and the quench spray and post-LOCA sump pH have been evaluated and found to be acceptable. The acceptance criteria of all non-LOCA transients continue to be met. Therefore, there is no significant reduction in the margin of safety in the accident analyses impacted by boron concentration increases in the RWST, CCT, SIAs, and SFP.

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

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

NRC Section Chief: L. Raghavan, Acting.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: June 22, 2000.

Description of amendment request: The proposed changes modify the limiting conditions for operation, surveillance requirements, and the Bases for the North Anna Power Station (NAPS) Units 1 and 2 Technical Specifications 3.4.1.4, 3.4.1.6, 4.4.1.4, 4.4.1.6.1, and add 4.4.1.6.4 to extend the drained reactor coolant loop verification time from 2 hours to 4 hours prior to backfilling when returning the drained loop to service. This amendment request supersedes the August 4, 1999, request in its entirety (64 FR 48868, September 8, 1999).

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:

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed [revision to the] Technical Specification[s] establishes limiting conditions for operation and surveillance requirements for isolated loops backfill. Specifically, Technical Specifications requirements are being established to control the source of borated water for seal injection to the reactor coolant pumps (RCP) and to address reactivity control of an isolated and filled loop. The proposed controls ensure that the boron concentration of any source of water used for reactor coolant pump seal injection is greater than or equal to the boron concentration corresponding to the shutdown margin requirements for the applicable Mode. The proposed changes will establish consistent reactivity controls for isolated Reactor Coolant Systems (RCS) loops. The Bases [have] been revised to further discuss the additional controls for the loop backfill evolution. Adequate Technical Specifications controls have been established to ensure that an uncontrolled positive reactivity addition does not occur during a loop backfill evolution. The proposed changes will ensure that an inadvertent/undetected positive reactivity addition does not occur. The following is provided to support this conclusion.

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

The proposed Technical Specification limiting conditions for operation and surveillance requirements ensure that the initiation of seal injection in order to allow a partial vacuum to be established in an isolated and drained loop will not create the potential for an inadvertent/undetected introduction of under-borated water into an isolated loop prior to returning the isolated loop to service. The proposed Technical Specification controls prevent any additions of makeup or seal injection that would violate the existing shutdown margin requirements for the active portion of the RCS. Thus, adequate Technical Specification controls are established to preclude an inadvertent/undetected boron dilution event. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

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

There are no modifications to the plant as a result of the changes. The proposed Technical Specification Limiting Conditions for Operation and Surveillance Requirements ensure that the initiation of seal injection will not create an undetected positive reactivity addition. No new accident or event initiators are created by the initiation of seal injection for the RCP in the isolated loop in order to establish a partial vacuum in that isolated and drained loop. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Involve a significant reduction in the margin of safety as defined in the bases on any Technical Specifications.

The proposed changes have no effect on safety analyses assumptions. Rather, the proposed changes acknowledge the establishment of seal injection for the RCP in the isolated and drained loop as a prerequisite for the vacuum-assisted backfill technique. The proposed Technical Specification Limiting Conditions for Operation and Surveillance Requirements ensure that the initiation of seal injection in order to allow a partial vacuum to be established in an isolated and drained loop will not create the potential for an inadvertent/undetected introduction of under-borated water into an isolated loop prior to returning the isolated loop to service. Adequate Technical Specifications controls are established to preclude an inadvertent/undetected boron dilution event. In addition, the proposed controls prevent any additions of makeup or seal injection that would violate the existing shutdown margin requirements for the active portion of the Reactor Coolant System. Therefore, the proposed changes do not result in 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: June 27, 2000 (WM 00–0026).

Description of amendment request: The proposed amendment would revise Appendix C, ''Antitrust Conditions for Kansas Gas and Electric Company [KGE],'' for the Wolf Creek Generating Station (WCGS) operating license. The revisions would (1) state that the specific conditions applicable to Kansas Electric Power Cooperative, Inc. (KEPCo) do not restrict its rights, or the duties of KGE, under other license conditions, (2) define ''KGE members in licensee's service area'' in the appendix to include all KEPCo members with facilities in Western Resources' and KGE's combined service area, (3) delete license conditions restricting KEPCo's use of the power from WCGS, (4) remove out-of-date conditions, and (5) update conditions to be consistent with the terms and conditions of Western Resources' Federal Energy Regulatory Commission (FERC) open access transmission tariff. Western Resources is the parent company of KGE.

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 merely revises the KGE Antitrust Conditions in the Wolf Creek Generating Station Facility Operating License. The proposed change is considered an administrative change and does not modify, add, delete, or relocate any technical requirements of the Technical Specifications. As such, the administrative changes do not affect 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 analyzed.

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 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 [requirements] or eliminate any old requirements. Thus, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because there is no effect on any safety analyses assumptions. The changes are administrative in nature. Therefore, the change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

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

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

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

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of application for amendment: June 7, 2000.

Brief description of amendment: The proposed amendment would revise Section 3.10.8, ''SHUTDOWN MARGIN (SDM) Test—Refueling,'' of the Technical Specifications (TS), correcting an administrative error introduced when Amendment No. 91 (converting the TS to the Improved TS format) was processed.

*Date of publication of individual notice in **Federal Register:*** June 16, 2000 (65 FR 37807).

Expiration date of individual notice: July 17, 2000.

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: May 26, 1999, as supplemented March 31, 2000.

Brief description of amendments: The amendments revise Technical Specification 3.3.1, ''Reactor Protective System (RPS) Instrumentation—Operating,'' to change the allowable values for two of the trip setpoints. The change will reduce spurious reactor trip hazards associated with these setpoints while maintaining plant protection.

Date of issuance: July 6, 2000.

Effective date: July 6, 2000, to be implemented within 60 days. For surveillance requirements associated with the revised allowable values for functions 12 and 13 in Technical Specification Table 3.3.1–1, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of these amendments.

Amendment Nos.: Unit 1–126, Unit 2–126, Unit 3–126.

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

Date of initial notice in **Federal Register:** May 17, 2000 (65 FR 31355).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2000.

No significant hazards consideration comments received: No.

Baltimore Gas and Electric Company, Docket Nos. 50–317, 50–318, and 72–8, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, and Independent Spent Fuel Storage Installation, Calvert County, Maryland

Date of application for amendment: February 29, 2000, as supplemented April 7, April 27, May 2, May 19, and June 20, 2000.

Brief description of amendment: These amendments conform the licenses to reflect the transfer of Operating Licenses Nos. DPR–53 and DPR–69 for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and Materials License No. SNM–2505 for the Calvert Cliffs Independent Spent Fuel Storage Installation held by Baltimore Gas and Electric Company to Calvert Cliffs Nuclear Power Plant, Inc.

Date of Issuance: June 30, 2000.

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

Amendment No.: 237 and 211.

Facility Operating License No. DPR–53, DPR–69: Amendments revised the Operating Licenses, and Materials License No. SNM–2505 and the Materials License Technical Specifications.

Date of initial notice in ***Federal Register:*** May 4, 2000 (65 FR 25963)

The April 7, April 27, May 2, May 19, and June 20, 2000, supplements did not expand the scope of the initial application as originally noticed.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated June 30, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 29, 1999.

Brief description of amendment: The amendment revises items 1.a, 2.a, 4.a, and 5.a of Technical Specification Table 3.3.5.1–1, ''Emergency Core Cooling System Instrumentation,'' to change the reactor vessel water level—level 1 allowable value.

Date of issuance: July 13, 2000.

Effective date: July 13, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 166.

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

Date of initial notice in ***Federal Register:*** August 25, 1999 (64 FR 46431) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 13, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 20, 1999.

Brief description of amendment: Incorporates 16 improvements (identified by Technical Specifications Task Force (TSTF) numbers) to the Improved Standard Technical Specifications, NUREG–1434 (for General Electric model Boiling Water Reactor/6 (BWR/6) plants such as Grand Gulf Nuclear Station (GGNS)), that was part of the basis for the current improved Technical Specifications for GGNS that were issued in Amendment 120 dated February 21, 1995. The 17 improvements are the following TSFTs: 2, 5, 17, 18, 32, 33, 38, 45, 60, 104, 118, 153, 163, 166, 278, and 279. The licensee withdrew its request to incorporate TSTF–9 into the TSs.

Date of issuance: June 30, 2000.

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

Amendment No.: 142.

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

Date of initial notice in ***Federal Register:*** December 29, 1999 (64 FR 73089).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 30, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 18, 1999, as supplemented by letters dated May 16, 2000, and June 1, 2000.

Brief description of amendment: The amendment modifies Technical Specification (TS) 3.6.2.2 Limiting Condition for Operation to allow Waterford Steam Electric Station, Unit 3 to operate with two independent trains of containment cooling, consisting of one cooler per train, operable during modes 1, 2, 3, and 4. Associated changes to the TS Bases have been incorporated.

Date of issuance: July 6, 2000.

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

Amendment No.: 165.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 9, 2000 (65 FR 6407). The May 16, 2000, and June 1, 2000, supplements did not expand the scope of the application as noticed or change the proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating 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: July 15, 1999.

Brief description of amendments: The amendments change the Technical Specifications (TSs) surveillance frequency for the quench and recirculation spray system nozzle air flow test. The amendments also change terminology in the TS action statement for the TS axial flux difference, and make other miscellaneous editorial and format changes.

Date of issuance: July 11, 2000.

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

Amendment Nos.: 231 and 111.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 17, 1999 (64 FR 62708) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 11, 2000.

No significant hazards consideration comments received: No.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: January 27, 2000, as supplemented May 30, 2000.

Brief description of amendment: The amendment will modify the action statement for Technical Specification (TS) 3/4.7.11, ''Ultimate Heat Sink,'' to permit Unit 2 to remain in operation with the ultimate heat sink water temperature greater than 75° F and less than 77° F, for a period of up to 12 hours provided the water temperature is verified below 77° F at least once per hour. This is a one-time change during the summer period and will expire after October 15, 2000, and revert back to the original TS action statement.

Date of issuance: July 10, 2000.

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

Amendment No.: 247.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15382) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 10, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: March 19, 1999.

Brief description of amendments: The amendments revise paragraph 2.C.(4) of the Operating Licenses related to the fire protection program at Prairie Island, Units 1 and 2. Specifically, the proposed amendments would (1) remove reference to two NRC safety evaluation reports (SEs) that are no longer applicable to the fire protection program at Prairie Island and (2) correct the date of one SE.

Date of issuance: July 11, 2000.

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

Amendment Nos.: 150 and 141.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 28, 2000 (65 FR 25001).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 11, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: April 12, 1999, as supplemented July 7, 2000.

Brief description of amendments: The amendments revise several Technical Specification (TS) sections in order to relocate shutdown margin requirements to the Core Operating Limits Report and to ensure that the TS requirements are consistent with the dilution analysis in the Updated Safety Analysis Report.

Date of issuance: July 11, 2000.

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

Amendment Nos.: 151 and 142.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 28, 2000 (65 FR 24999). The July 7, 2000, supplemental letter provided clarifying information that was within the scope of the original application and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 11, 2000.

No significant hazards consideration comments received: No.

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Units 1 and 2, Goodhue County, Minnesota

Date of application for amendments: November 17, 1999, as supplemented April 6, 2000.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.1.A.1, ''Reactor Coolant Loops and Coolant Circulation,'' to (1) establish required actions and a 72 hour time limit for operation with the reactor coolant system (RCS) average temperature above 350 °F and no reactor coolant pumps (RCPs) running, (2) extend from 6 hours to 12 hours the time within which the RCS average temperature must be reduced to below 350 °F if 72 hours are exceeded and no RCPs are restored to operability and operation, and (3) extend the time limit for operations with no RCPs running from 1 hour to 12 hours for situations where the RCPs are stopped as a result of preplanned work activities.

Date of issuance: July 14, 2000

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

Amendment Nos.: 152 and 143

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 29, 1999 (64 FR 66670).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 14, 2000.

No significant hazards consideration comments received: No.

PP&L, Inc., Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: December 15, 1999, as supplemented February 7, March 24, April 28, May 4, and May 30, 2000.

Brief description of amendments: The amendments conform the operating licenses for each of the units to reflect the transfer of the operating licenses, to the extent held by PP&L, Inc., to PPL Susquehanna, LLC.

Date of issuance: July 1, 2000

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

Amendment Nos.: 188 and 162.

Facility Operating License Nos. NPF–14 and NPF–22. The amendments revised the license.

Date of initial notice in ***Federal Register:*** March 3, 2000 (65 FR 11611). The March 24, April 28, May 4, and May 30, 2000, letters provided clarifying information. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 6, 2000.

Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: April 14, 1999, as supplemented on March 2, 2000.

Brief description of amendments: The license amendment revises Technical Specification (TS) Section 3/4.9.12, ''Fuel Handling Area Ventilation System,'' and provides greater consistency between the two Salem units, removes inappropriate and invalid surveillance requirements (SR), and clarifies the Bases. The revised TS Section 3/4.9.12 will require that the high efficiency particulate air (HEPA) and charcoal filters to be in service prior to moving irradiated fuel in the Fuel Handling Building. This will be accomplished by the addition of a new SR 4.9.12.b. The new SR allows the licensee to eliminate an automatic actuation feature from the Fuel Handling Area Ventilation system control circuit, as well as the requirement to test that feature. The new surveillance will also require verification of system line up every 24 hours during fuel movement or crane operation to ensure system flow through the HEPA-charcoal filter train.

Date of issuance: June 14, 2000.

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

Amendment Nos.: 231 & 211.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 2, 1999 (64 FR 29715). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 14, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 17, 1999.

Brief description of amendment: The proposed changes would revise the required minimum contained volume of the condensate storage tank from 172,700 gallons of water to 179,850 gallons of water.

Date of issuance: July 7, 2000.

Effective date: July 7, 2000.

Amendment No.: 145.

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

*Date of initial notice in **Federal Register:*** June 16, 1999 (64 FR 32290).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 7, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 2, 1999, as supplemented May 16 and June 16, 2000 (PCN–506).

Brief description of amendments: These amendments approve changes to Technical Specifications, Section 5.0, ''Administrative Controls,'' and the Environmental Protection Plan.

Date of issuance: July 7, 2000.

Effective date: July 7, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2–168; Unit 3–159.

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

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73096). The May 16 and June 16, 2000, letters provided additional information and clarifications that were within the scope of the original **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 July 7, 2000.

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: October 18, 1999, as supplemented May 11, 2000.

Brief description of amendment: The amendment revises the Technical Specifications to require a revised activated charcoal testing methodology in accordance with the guidance provided by Generic Letter 99–02, ''Laboratory Testing of Nuclear Grade Activated Charcoal.''

Date of Issuance: July 11, 2000.

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

Amendment No.: 189.

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

*Date of initial notice in **Federal Register:*** November 17, 1999 (64 FR 62716).

The May 11, 2000, supplement did not expand the scope of the application as initially noticed, or change the proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 11, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 19th day of July 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–18771 Filed 7–25–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[NUREG–1620]

Review of A Reclamation Plan For Mill Tailings Sites Under Title II of the Uranium Mill Tailings Radiation Control Act; Final Standard Review Plan

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has published the Final Standard Review Plan for Review of a Reclamation Plan for Mill Tailings Sites Under Title II of the Uranium Mill Tailings Radiation Control Act (NUREG–1620). An NRC source and byproduct material license is required under the provisions of Title 10 of the Code of Federal Regulations, part 40 (10 CFR part 40), Domestic Licensing of Source Material, in conjunction with uranium or thorium milling, or with byproduct material at sites formerly associated with such milling. An applicant for a new reclamation plan, or for the renewal or amendment of an existing license, is required to provide detailed information on the facilities, and procedures to be used, and if appropriate, an environmental report that discusses the effect of proposed operations on public health and safety and on the environment. This information is used by Nuclear Regulatory Commission staff to determine whether the proposed activities will be protective of public health and safety and the environment. The standard review plan provides guidance to NRC staff for the review of reclamation plans while ensuring consistency and uniformity among the staff reviews. Each section in the review plan provides detailed review guidance on subject matter required in a standard reclamation plan. The review plan is intended to improve the understanding of the staff review process by interested members of the public and the uranium recovery industry. The final version includes updates based on public comment on the draft Standard Review Plan for the Review of a Reclamation Plan for Mill Tailings Sites Under Title II of the Uranium Mill Tailings Radiation Control Act.

Availability: Copies of NUREG–1620 may be purchased from the Superintendent of Documents, U.S. Government Printing Office, PO Box 37082, Washington, DC 20402–9328. Copies are also available from the National Technical Information Service, 5285 Port Royal Road, Springfield, Virginia 22161. Paper and electronic copies are available for inspection and/or copying in the NRC Public Document Room, 2120 L Street, NW. Washington, DC. An electronic copy can be accessed for reading, searching, or copying under ''Technical Reports in the NUREG Series'' of the ''NRC Reference Library'' at the NRC Web site, (http://www.nrc.gov/NRC/NUREGS).

Dated at Rockville, Maryland, this 3rd day of July, 2000.

For the Nuclear Regulatory Commission.

Philip Ting,

Chief, Fuel Cycle Licensing Branch, Division of Fuel Cycle Safety and Safeguards Office of Nuclear Material Safety and Safeguards.

[FR Doc. 00–18919 Filed 7–25–00; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Cumulative Report on Rescissions and Deferrals

July 1, 2000.

Section 1014(e) of the Congressional Budget and Impoundment Control Act of 1974 (Public Law 93–344) requires a monthly report listing all budget authority for the current fiscal year for which, as of the first day of the month, a special message had been transmitted to Congress.

This report gives the status, as of July 1, 2000, of three rescission proposals and two deferrals contained in one special message for FY 2000. The message was transmitted to Congress on February 9, 2000.