amendment is effective as of the date of issuance.

The amendment modified the Operating License and TS to allow an increase of the authorized operating power level from 1658 megawatts thermal (MWt) to 1912 MWt at DAEC. The change represents an increase of 15.3 percent above the current rated thermal power and is considered an extended power uprate.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the **Federal Register** on September 27, 2001 (66 FR 49426). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (66 FR 55703).

Further details with respect to the action see (1) the application for amendment dated November 16, 2000, as supplemented April 16 (two letters) and 17; May 8 (two letters), 10, 11 (two letters), 22, and 29; June 5, 11, 18, 21, and 28; July 11, 19, and 25; August 1, 10, 16, and 21; and October 17, 2001, (2) Amendment No. 243 to License No. DPR–49, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC Public Document Room Reference staff by telephone at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 6th day of November 2001.

For the Nuclear Regulatory Commission.

Brenda L. Mozafari,

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

[FR Doc. 01–28510 Filed 11–13–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Meetings; Sunshine Act

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of November 12, 19, 26, December 3, 10, 17, 2001.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of November 12, 2001

Wednesday, November 14, 2001

8:55 a.m.—Affirmation Session (Public Meeting) (if needed)

9:00 a.m.—Discussion of Intragovernmental and Security Issues (Closed-Ex. 1 & 9)

Thursday, November 15, 2001

2:00 p.m.—Discussion of Intragovernmental Issues (Closed-Ex. 1)

Week of November 19, 2001—Tentative

There are no meetings scheduled for the Week of November 19, 2001.

Week of November 26, 2001—Tentative

There are no meetings scheduled for the Week of November 26, 2001.

Week of December 3, 2001—Tentative

Monday, December 3, 2001

2:00 p.m.—Briefing on Status of Steam Generator Action Plan (Public Meeting) (Contact: Maitri Banerjee, 301–415–2277)

Wednesday, December 5, 2001

1:25 p.m.—Affirmation Session (Public Meeting) (if needed)

1:30 p.m.—Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360)

Week of December 10, 2001—Tentative

There are no meetings scheduled for the Week of December 10, 2001.

Week of December 17, 2001—Tentative

There are no meetings scheduled for the Week of December 10, 2001.

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

Additional Information: By a vote of 5–0 on November 2, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Intragovernmental and Security Issues (Closed-Ex. 1 & 9)'' be held on November 6, and on less than one week's notice to the public.

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

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

Dated: November 8, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–28644 Filed 11–9–01; 2:19 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

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 October 22, 2001 through November 3, 2001. The last biweekly notice was published on October 31, 2001 (66 FR 557007).

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

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

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

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

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

By December 14, 2001, 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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to *pdr@nrc.gov.*

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

Date of amendment request: August 21, 2001.

Description of amendment request: The proposed amendment would revise the actions taken for an inoperable battery charger, revise battery charger testing criteria, and relocate certain safety-related battery surveillance requirements from the Technical Specifications to a licensee-controlled 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:

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

The proposed changes restructure the TS [Technical Specifications] for the DC Electrical Power system. The proposed changes add actions to specifically address battery charger inoperability with increased completion times. This change will rely upon the capability of providing the battery charger function by an alternate means, (e.g., a spare battery changer that will function as a qualified backup) to take advantage of the proposed increased completion time. The CD power System or associated battery chargers are not initiators to any accident sequence analyzed in the Updated Safety Analysis Report (USAR). Operation in accordance with the proposed TS ensures that the DC Power System is capable of performing function as described in the USAR, therefore the mitigative functions supported by the DC Power System will continue to provide the protection assumed by the analysis.

The relocation of preventive maintenance surveillance, and certain operating limits and actions to a newly-created, licensee-controlled TS 5.5.14, ''Battery Monitoring and Maintenance Program,'' will not challenge the ability of the DC Power System to perform its design function. The maintenance and monitoring required by current TS, which are based on industry standards, will continue to be performed. In addition, the DC Power System is within the scope of 10 CFR 50.65, ''Requirements for monitoring the effectiveness of maintenance at nuclear power plants,'' which will ensure the control of maintenance activities associated with the DC Power System.

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

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

The proposed changes involve restructuring the TS for the DC Electrical Power system. This change will rely upon the capability of providing the battery charger function by an alternate means, (*e.g.,* a spare battery charger that will function as a qualified backup) to take advantage of the proposed increased completion time. The DC Power System or associated battery chargers are not initiators to any accident sequence analyzed in the Updated Safety Analysis Report (USAR).

Allowing the use of a spare battery charger will increase the reliability of the DC Electrical Power system. The mitigative functions supported by the DC Power System will continue to provide the protection assumed by the safety analysis described in the USAR. Therefore, there are no new types of failures that could be created by a failure of the spare battery charger. As such, no new or different kind of accident or transient is expected by these changes.

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

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

The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The changes associated with the new Battery Maintenance and Monitoring Program will ensure that the station batteries are maintained in a highly reliable manner. The use of a spare battery charger will increased the reliability of the DC system during periods of normal battery charger inoperability. The equipment fed by the DC Electrical Sources will continue to provide adequate power to safety related loads in accordance with analysis assumptions.

Therefore, the proposed 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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: September 20, 2000, as supplemented August 2 and September 28, 2001.

Description of amendment request: The proposed Technical Specification (TS) change would (1) delete the requirements for hydrogen monitoring instrumentation from TS sections 3.5.5.2, 3.6, and Tables 3.5–3 and 4.1–4 and correct a typographical error in item 8 of Table 4.1–4; (2) delete the requirements for hydrogen recombiners in TS section 4.4.4; and (3) delete the reference to the hydrogen purge system and hydrogen recombiners from the Bases of TS section 4.12.2.

Basis for proposed no significanthazards 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 Nuclear Regulatory Commission (NRC) staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representation made by the licensee in the September 20, 2001, application as supplemented August 2 and September 28, 2001, is presented below:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. This change has no effect on plant equipment provided for the reactor coolant system, reactor building heat removal, or the equipment provided for mixing of the reactor building atmosphere following an accident. This proposed change does not alter the design or configuration of the plant beyond that of the containment combustible gas control systems. The containment combustible gas control systems are currently classified as safety systems. The containment combustible gas control systems are composed of two hydrogen monitors and two hydrogen recombiners, backed up by a portion of the reactor building purge system that can be used to vent the reactor building. Hydrogen control components (hydrogen monitors, hydrogen recombiners, and hydrogen vents) do not affect any accident initiation sequence previously identified. Therefore, this change does not increase the probability of an accident previously evaluated.

The containment combustible gas control systems are provided to ensure that reactor building hydrogen concentration is maintained below the lower flammability limit of 4.0 percent. The NRC staff has found hydrogen combustion to be a small contributor to containment failure for large, dry containment designs due to the robustness of these containment types and the likelihood of a spurious ignition source. The containment combustible gas control systems are not credited in the TMI Unit 1 probability risk assessment (PRA).

Therefore, this change would not result in a significant increase the consequence of accidents previously evaluated.

2. Will operation of the facility in accordance with the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

No. This proposed change does not alter the design or configuration of the plant beyond that of the containment combustible gas control systems. Hydrogen generation following a design basis loss-of-coolant accident (LOCA) has been evaluated in accordance with regulatory requirements. Deletion of the containment combustible gas control system from the TSs does not alter the hydrogen generation processes post-LOCA. The NRC staff has found hydrogen combustion to be a small contributor to containment failure for large, dry containment designs due to the robustness of these containment types and the likelihood of a spurious ignition source. The containment combustible gas control systems are not credited in the TMI Unit 1 level 2 PRA.

Therefore, since the accident evaluation does not credit these systems or assume that they operate during an accident, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

No. This change has no effect on plant equipment provided for the reactor coolant system, reactor building heat removal, or the equipment provided for mixing of the reactor building atmosphere following an accident. This change only involves the deletion of requirements for containment combustible gas control equipment, (hydrogen monitors, hydrogen recombiners, and containment hydrogen vents). The NRC staff has found hydrogen combustion to be a small contributor to containment failure for large, dry containment designs due to the robustness of these containment types and the likelihood of a spurious ignition source. Use of the containment combustible gas control systems are not credited in the TMI Unit 1 PRA. TMI Unit 1 utilizes a large open containment design that precludes the buildup of hydrogen pockets that might be formed if the reactor building were of a compartmentalized design. The TMI–1 PRA concluded that the containment would remain intact for severe accidents which included hydrogen burns for which no credit was taken for the combustible gas control system as long as the containment heat removal systems (reactor building emergency cooling and reactor building sprays) remain functional.

The proposed change will relax certain special treatment requirements associated with hydrogen monitors. As discussed in Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,'' Revision 3, dated May 1983, the NRC staff believes that the revised treatment is appropriate for instrumentation needed to assess the degree of core damage and confirm that spurious ignition of hydrogen has taken place.

Therefore, operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Edward J. Cullen, Jr., Vice President and General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, KSB 3-W, Kennett Square, PA 19348.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: August 27, 2001.

Description of amendment request: The proposed amendment changes the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications (TSs) action and surveillance requirements associated with the containment air lock. The Bases of the affected TSs will be modified to address the proposed changes.

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 reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representation made by the licensee in the August 27, 2001, application, is presented below:

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

The proposed changes will not revise the operability requirements for the containment air lock. As a result, the design-basis accidents will remain the same postulated events, and the consequences of the design-basis accidents will remain the same. Also, the containment air lock is not an accident initiator. Therefore, the proposed change will not involve any 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.

Since the containment air lock is not an accident initiator, these proposed changes do not introduce any new failure modes. 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.

Since the operability requirements for the containment air lock will not change, and the containment air lock will continue to function as assumed in the safety analysis, the proposed change will not result in a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 26, 2001.

Description of amendment request: The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications (TSs) to relocate MP3 TSs related to the position indication system to the respective Technical Requirements Manual (TRM). The Bases of the affected TSs will be modified to address the proposed changes. Also, index pages will be revised to reflect the relocation.

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 reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representation made by the licensee in the September 26, 2001, application, is presented below:

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

The proposed requirements remain the same except that the requirements will be relocated to the TRM. Since the proposed requirements are the same, this proposed change will not 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.

Since the requirements remain the same, these proposed changes do not alter the way any system, structure, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. 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.

Since the proposed changes are solely to relocate the existing requirements, it does not affect plant operation in any way. Therefore, the proposed change will not result in a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 1, 2001.

Description of amendment request: The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications (TS) to change TS 3.4.6.2 ''Reactor Coolant System—Operational Leakage''. The Bases for this TS will also be modified to reflect this 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, which is presented below:

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

The proposed changes to Technical Specification 3.4.6.2 for [reactor coolant systems] RCS PIVs [pressure isolation valve] in the RHR [residual heat removal] flow path will not cause an accident to occur and will not result in any change in the operation of associated accident mitigation equipment. The ability of the RHR System to remove core decay heat will not be affected. The proposed changes will not affect the ability of the RCS or the RHR System to mitigate any design basis event. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR), and the consequences of the design basis accidents will remain the same. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed changes to delete SRs 4.4.6.2.1.a and 4.4.6.2.1.b and revise SR [Surveillance Requirement] 4.4.6.2.1.d will not cause an accident to occur and will not result in any change in the operation of associated accident mitigation equipment. The ability to measure RCS operational leakage will not be affected. The proposed changes will not affect the ability to mitigate any design basis event. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 FSAR, and the consequences of the design basis accidents will remain the same. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The proposed change to remove SR 4.4.6.2.2.c to perform post maintenance testing of the RCS PIVs will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The proposed change will not revise the operability requirements (*e.g.,* valve leakage limits) for the RCS PIVs. Proper operation of the RCS PIVs will still be verified, as appropriate, following maintenance activities. As a result, the design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 FSAR, and the consequences of the design basis accidents will remain the same. Therefore, the proposed change will not 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 do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce the margin of safety since they have no impact on any accident analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions. The effectiveness of Technical Specifications will be maintained since the changes will not alter the operation of any component or system, nor will the proposed changes affect any safety limits or safety system settings. Therefore, there is no 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, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: August 22, 2001.

Description of amendment request: The proposed amendment would change the Technical Specification (TS) Surveillance Requirement 3/4.7.B.1.a.2 for the Standby Gas Treatment (SBGT) System by increasing the SBGT inlet heaters minimum output testing requirement from 14 kW to 20 kW. The associated TS Bases 3/4.7.B.1 would also be revised as a result of the proposed TS change. The proposed change is based upon the licensee's revised design-basis calculations for the SBGT inlet heaters and by a modification that replaces the existing SBGT system inlet heaters with heaters of higher output capability.

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

The proposed change affects only the surveillance requirement for the SBGT inlet heaters output capability. The SBGT heaters are not the initiators of any accidents described in the safety analysis report (SAR). The proposed higher inlet heater output capability test is needed to ensure that the SBGT will continue to function as currently designed to decrease the relative humidity (RH) of the inlet air stream to 70% RH. The higher inlet heater output capability test does not change the consequences of an accident previously analyzed in the SAR. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to the SBGT inlet heaters capacity surveillance testing requirement is needed to continue to ensure that the SBGT will function to decrease the RH of the inlet air stream to 70% RH, as assumed in the current analysis. The SBGT heaters are not the initiators of any accidents described in the SAR. The proposed change in the surveillance testing requirement does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed higher testing acceptance criteria for the inlet heater ensures that the SBGT will continue to function as currently designed to decrease the RH of the inlet air steam to 70% RH. The margin of safety is unaffected by this change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 17, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) actions regarding inoperable redundant components when an Emergency Diesel Generator (EDG) becomes inoperable. TS 3.8.1.1 would be revised to require actions based on the TS for the inoperable redundant component(s). The proposed revision is consistent with NUREG–1432, Rev.2, ''Standard Technical Specifications, Combustion Engineering Plant.''

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

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

Neither the steam driven auxiliary feedwater pump nor the EDGs are accident initiators, but are accident mitigators. The proposed changes to the EDG TS do not affect the operation nor availability of the EDGs, the motor or steam driven auxiliary feedwater pumps, nor TS required redundant features. For those conditions that would require a unit shutdown, once the four hour completion time had expired, the shutdown would be performed in the manner and timeframe supported by the existing redundant feature TS. Therefore, the probability or consequences of any accident previously evaluated have not been significantly increased.

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

No new failure modes are introduced by the proposed TS changes and single failure considerations are adequately addressed by following the established conventions of NUREG–1432. The proposed four hour completion time from the discovery of inoperable redundant features and an EDG takes into account the operability of the redundant counterpart to the inoperable required feature, the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA [design-basis accident] occurring during this period. The TS change required reformatting and moving the steam driven auxiliary feedwater pump operability requirements to the redundant feature(s) actions to be comparable with and meet the intent of the BASES requirements contained in NUREG–1432. Without creation of a new interaction of materials, operating configuration, or operating interfaces, there is no possibility that the proposed changes can introduce a new or different kind of accident.

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

The margin of safety as defined in the basis for any Technical Specification or in any licensing document has not been reduced. The proposed changes remove the unconditional unit shutdown requirement should an EDG be inoperable while required features on the opposite train are inoperable. Instead, any TS required actions are appropriately based on the inoperability of the required feature. The proposed four hour completion time from the discovery of inoperable redundant features and an EDG takes into account the operability of the redundant counterpart to the inoperable required feature, the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. For those conditions that would require a unit shutdown, once the four hour completion time had expired, the shutdown would be performed in the manner and timeframe supported by the existing redundant feature TS. Additionally, the TS requirements to assure that steam driven auxiliary feedwater pump operability is considered as part of the redundant features requirements remains and is comparable to the intent of the BASES of STS 3.8.1. Based on the preceding discussion, FPL concludes that the margin of safety will not be significantly reduced by operation of the facility in accordance with the proposed amendments.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: October 17, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) multiplier values for single-loop operation (SLO) average planar linear heat generation rate (APLHGR).

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 proposed APLHGR multipliers, and their use to determine the Cycle 21 thermal limits, have been derived using NRC approved methods and uncertainties. These methods do not change operation of the plant, and have no effect on the probability of an accident initiating event or transient. The purpose of the APLHGR limit is to assure that the fuel will not exceed a peak cladding temperature (PCT) of 2200 °F during a Loss of Coolant Accident [LOCA], as required by 10 CFR 50.46. Specifying appropriate APLHGR multipliers ensures that a LOCA in SLO will not produce a PCT any greater than the PCT produced by a LOCA in dual loop operation. These changes ensure that the appropriate SLO APLHGR multiplier, required for GE14 fuel, is incorporated into the Monticello TS. These changes do not alter the method of operating the plant.

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

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

The proposed changes result only from different inputs, including use of GE14 fuel, for the Cycle 21 core reload. These methods and uncertainties have been reviewed and approved by the NRC, and do not involve any new or unapproved methods for operating the facility. No new initiating events or transients result from these changes.

The single-loop operation APLHGR multiplier values are designed to ensure that the PCT resulting from a LOCA while operating in SLO are bounded by the PCT from a LOCA while operating in dual loop operation. This multiplier update results from application of GE Nuclear Energy's (GE's) current standard methodology for this analysis.

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

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

The APLHGR limits are set appropriately below the value where significant fuel damage could occur in a Loss of Coolant Accident (LOCA). Application of new SLO APLHGR multiplier values ensure that SLO LOCA results are bounded by those for dual loop operation and thus maintain or improve the margin of safety for LOCA analyses.

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

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

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

NRC Section Chief: William D. Reckley.

PPL Susquehanna, LLC, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: September 19, 2001.

Description of amendment request: The proposed amendment would revise the Unit 1 reactor pressure vessel (RPV) material surveillance program to defer the withdrawal of the second surveillance capsule for one operating cycle. Deferral is requested to support PPL Susquehanna, LLC's, participation in the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance 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:

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

Pressure-temperature (P/T) limits are imposed on the reactor coolant system to ensure that adequate safety margins against non-ductile or rapidly propagating failure exist during normal operation, anticipated operational occurrences, and system hydrostatic tests. The P/T limits are related to the nil-ductility reference temperature, RTndt. Changes in the fracture toughness properties of the Reactor Pressure Vessel (RPV) beltline materials, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of 10 CFR 50, Appendix H. The effect of neutron fluence on the shift in the nil-ductility reference temperature of pressure vessel steel is predicted by methods given in Regulatory Guide (RG) 1.99, Revision 2 and Regulatory Guide 1.190, Revision 0. The Susquehanna SES [Steam Electric Station] Unit 1 current P/T limits were established based on adjusted reference temperatures developed in accordance with the procedures prescribed in RG 1.99, Revision 2. Calculation of adjusted reference temperature by these procedures includes a margin term to ensure upperbound values are used for the calculation of the P/T limits. Revision of the second capsule withdrawal schedule will not affect the P/T limits, because they will continue to be established in accordance with NRC approved methodology in accordance with RG 1.190 Revision 0 commitments. The existing P/T limits are based on 32 EFPY rather than for the planned withdrawal at 15 EFPY. This change is not related to any accidents previously evaluated. The proposed change will not affect reactor pressure vessel performance because no physical changes are involved and the RPV vessel P/T limits will remain in accordance with RG 1.99, Revision 2 commitments. The proposed change will not cause the reactor pressure vessel or interfacing safety systems to be operated outside of their design or testing limits. Also, the proposed change will not alter any assumptions previously made in evaluating the radiological consequences of accidents.

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

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

The proposed change defers the second RPV material surveillance capsule withdrawal for one fuel cycle. This proposed change does not involve a modification of the design of plant structures, systems, or components. The proposed change will not impact the manner in which the plant is operated as plant operating and testing procedures will not be affected by the change. The proposed change will not degrade the reliability of structures, systems, or components important-to-safety because equipment protection features will not be deleted or modified, equipment redundancy or independence will not be reduced, supporting system performance will not be downgraded, the frequency of operation of equipment important-to-safety will not be increased, and more severe testing of equipment important-to-safety will not be imposed. No new accident types or failure modes will be introduced as a result of the proposed change.

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

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

Appendix G to 10 CFR 50 describes the conditions that require P/T limits and provide the general bases for these limits. Until the results from the reactor vessel surveillance program become available, RG 1.99, Revision 2 is used to predict the amount of neutron irradiation damage. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, provide reasonable assurance that nonductile or rapidly propagating failure will not occur. The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary (RCPB). Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition. The proposed change will not affect any safety limits, limiting safety system settings, or limiting conditions of operation. The proposed change does not represent a change in initial conditions, or in a system response time, or in any other parameter affecting the course of an accident analysis supporting the Bases of any Technical Specification. The proposed change does not involve revision of the P/T limits, but rather a revision of the withdrawal time for the second surveillance capsule. The current P/T limits were established based on adjusted reference temperatures for vessel beltline materials calculated in accordance with RG 1.99, Revision 2. P/T limits will continue to be revised, as necessary, for changes in adjusted reference temperature due to changes in fluence when two or three credible surveillance data sets become available. When two or more credible surveillance data sets become available, P/T limits will be revised as prescribed in RG 1.190, Revision 0.

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

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: July 30, 2001, as supplemented August 7, and October 16, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time deferral of the Type A containment integrated leakage rate test (ILRT) at the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The Unit 1 test would be deferred to no later than May 3, 2007, and the Unit 2 test would be deferred to no later than October 30, 2007, resulting in an extended interval of 15 years for performance of the next ILRT at each 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. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The frequency of Type A testing does not change the probability of an event that results in core damage or vessel failure. Primary containment is the engineered feature that contains the energy and fission products from evaluated events. The SSES IPE [Individual Plant Examination] documents events that lead to containment failure. The frequency of events that lead to containment failure does not change because it is not a function of the Type A test interval. Containment failure is a function of loss of safety systems that shutdown the reactor, provide adequate core cooling, provide decay heat removal, and drywell sprays.

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. Normally, extending a test interval increases the probability that a Structure System or Component will be failed. However, NUREG–1493, Performance-Based Containment Leak-Test Program, states that calculated risks in BWR's is very insensitive to the assumed leakage rates. The remaining testing and inspection programs provide the same coverage as the Type A test. These other programs will maintain containment leakage low. Any leakage path problems will be identified and repairs will be made. Additionally the containment is continuously monitored during power operation. Anomalies are investigated and resolved. Thus there is a high confidence that containment integrity will be maintained independent of the Type A test frequency.

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

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

Primary containment is designed to contain energy and fission products during and after an event. The SSES IPE identifies events that lead to containment failure. Revision to the Type A test interval does not change this list of events. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting mitigation of an accident.

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

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

The proposed revision to Technical Specifications adds a one time extension to the current interval for Type A testing. The current level of 10 years, based on past performance, would be extended on a one time basis to 15 years from the last Type A test. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1% to the individual risk and that increasing the Type A test interval would have minimal affect on this risk since 95% of the potential leakage paths are detected by Type B and Type C testing. Technical Specifications require that maximum allowable primary containment leakage rate is less than 1% primary containment air weight per day. During unit startup following Type B and Type C testing, leakage rate acceptance criteria must be less than 0.6% primary containment air weight per day. (TS 5.5.12) Therefore, Type B and Type C testing combined with visual inspection programs will maintain containment leakage low.

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

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: April 16, 2001, as supplemented on July 5, 2001.

Description of amendment request: The proposed Technical Specifications (TSs) change would modify required actions and surveillance requirements (SR) associated with the 28 Volt Direct Current (VDC) Battery System. The proposed changes are consistent with TS and SR requirements for the 125 VDC Battery System, and NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants.''

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

The proposed changes to TS limiting conditions for operation (LCOs) and surveillance requirements (SRs) will not alter the plant's physical configuration or the operation of the 28 VDC Battery System. As a result, the parameters assumed in the Salem Updated Final Safety Analysis Report (UFSAR) Design Basis Accident or Transient Analyses remain unchanged. Therefore, the probability or consequences of an accident previously evaluated are not increased by the proposed change.

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

The proposed changes to the 28 VDC Battery System TS LCOs and SRs do not modify the facility's design or physical configuration or change the method by which any safety-related system performs its function. Therefore, the proposed changes will not increase the possibility of a new or different kind of accident from any accident previously identified.

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

The proposed changes do not alter the manner in which safety limits or limiting safety system setpoints are determined. As a result, margins of safety are not changed. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 24, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.9, ''Refueling Operations,'' by relocating requirements for boron concentration to the Core Operating Limits Report (COLR). The proposed amendment will revise Limiting Condition for Operation (LCO) 3.9.1 by stating that, while the plant is in Mode 6, boron concentration of the Reactor Coolant System (RCS), refueling canal, and the refueling cavity shall be maintained within the limits specified in the COLR. LCO 3.9.1 required actions will also be revised to reference the COLR, and associated surveillance requirements will be changed to state that boron concentration shall be verified to be within the limits provided in the COLR every 72 hours.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. 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. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Relocating the minimum required boron concentration values from the TSs to the COLR does not change boron concentration requirements. Specifying the required minimum boron concentration in the COLR will continue to ensure that the proper boron concentration will be maintained in accordance with all the assumptions of appropriate accident analyses. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change relocates the minimum required boron concentration values from the TSs to the COLR. Moreover, the proposed change does not physically change the facility, plant operations, or the manner and frequency at which associated boron concentration testing is conducted. Therefore, the proposed change to relocate the required boron concentration to the COLR does not create the possibility of a new or different kind of accident from any previously analyzed.

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

Minimum boron concentration limits are established to ensure that sufficient margins exist to prevent criticality in the RCS, refueling canal, and the refueling cavity during refueling operations. Since the COLR is prepared as part of each core reload safety evaluation to ensure that current safety analysis limits are met, relocating the minimum boron concentration from the TSs to the COLR will not reduce safety margins. Therefore, the new proposed change to relocate the required boron concentration to the COLR does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: 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 19, 2001.

Description of amendment request: South Carolina Electric & Gas Company (SCE&G) proposes a change to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirements to revise Table 3.7–1. This change will identify maximum allowable power range neutron flux high setpoints based on the plant safety analysis or conservatively derived values calculated in accordance with NRC Information Notice 94–60 and Westinghouse Nuclear Safety Advisory Letter NSAL–94–001.

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

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

The proposed changes to Technical Specification 3.7.1 and its associated bases do not contribute to the initiation of any accident previously evaluated. Supporting factors are as follows:

All NSSS components are compatible with the revised core power limits and resulting operating conditions. Their structural integrity is maintained during all proposed plant conditions through compliance with the ASME code.

Other systems important to safety are not adversely impacted and will continue to perform their design functions.

The revised core power limits and resulting operating conditions remain within the design envelope of the plant.

Therefore, since the reactor coolant pressure boundary integrity and system functions are not adversely impacted, the probability of occurrence of an accident previously evaluated will be no greater than the existing design basis of the plant. The revised method to derive allowable power levels with inoperable main steam safety valves results in lower High Flux Trip Setpoints. When implemented, the revised trip setpoints ensure that secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transients. Since the ASME and regulatory limits on secondary side overpressurization will be met, the proposed changes will not create the potential for an increase in offsite releases or doses for any accident. Therefore, there is no increase in the consequences of an accident previously evaluated.

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

The proposed changes to Technical Specification 3.7.1 and its associated bases do not introduce any new accident initiator mechanisms. Structural integrity of the RCS and the secondary side is maintained during the allowed operating conditions, and ASME code limits continue to be met during all anticipated operating conditions. In addition, no new failure modes or limiting single failure or new design requirements for auxiliary systems are being introduced. Since the safety and design requirements continue to be met and the integrity of the primary and secondary pressure boundary is maintained, no new accident scenarios have been created. Therefore, the types of accidents previously defined continue to represent the credible spectrum of events to be analyzed. A new or different kind of accident is thus not created.

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

The proposed changes to Technical Specification 3.7.1 and its associated bases preserve the results and conclusions of plants safety analyses presented in the FSAR. The proposed changes address an identified deficiency with the current Technical Specification and, when implemented, restores the margin of safety intended. Specifically, the proposed changes ensures overpressure ensure that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. Therefore, there is no reduction in a margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: September 19, 2001.

Description of amendment request: The proposed amendments would revise surveillance requirement 3.6.1.3.8 which currently requires verification of the actuation capability of each reactor instrumentation excess flow check valve (EFCV) every 18 months. The proposed amendments would state that a representative sample of the EFCVs will be tested every 18 months such that each EFCV will be tested at least once every 10 years. The proposed amendments are consistent with Technical Specification Task Force-334.

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 a previously evaluated event?

The Excess Flow Check Valves are designed to limit the flow from an instrument line break downstream of the check valve itself. Thus the previously analyzed event is the instrument line break, documented in the Unit 2 FSAR, section 15.4.13, for both units. This proposed revision does not alter the operation or maintenance of any instrument lines; the revision is made to reduce the surveillance requirements for the EFCVs. This revision does nothing which jeopardizes the integrity of the instrument lines and thus increase the probability of a line break.

The line break analysis does not take credit for operation of the excess flow check valves, therefore, the radiological consequences of this event are not affected by this proposed TS revision.

This amendment request does not affect any other previously evaluated line or pipe break analsis.

For the above reasons, the probability of occurrence, or the consequences of a previously evaluated event are not increased by this proposed change.

2. Do the proposed changes create the possibility of a new type event different from any previously evaluated?

No changes are being made to the way in which the EFCVs are operated, or maintained; they will continue to be operated within the conditions for which they were designed. Since no new operational modes are proposed, no new failure modes are introduced.

Furthermore, no changes to any systems designed for the prevention of transients or accidents are being made as a result of this proposed Technical Specification change.

For the above reasons, this proposed change does not introduce the possibility of a different type event from any previously evaluated.

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

The reactor coolant pressure boundary line break analysis documented in Unit 2 FSAR section 15.4.13 does not assume credit for the EFCVs. Additionally, the failure rate of the Unit 1 and 2 EFCVs has been small, as verified by the failure rate analysis done for this proposed revision. Accordingly, reducing the frequency of the surveillance is justified and will not significantly reduce the margin of safety with respect to EFCV failure.

Additionally, General Electric has performed a generic radiological evaluation of an instrument line break, with EFCV failure, which concluded that the dose consequences would not exceed 10 CFR 100 guidelines. This analysis is documented in NEDO–32977–A, ''Excess Flow Check Valve relaxation'', a report commissioned by the Boiling Water Reactors Owners' Group (BWROG). Because the Hatch EFCV design is similar to the EFCV designs assumed in the NEDO, it is reasonable to conclude that the results of this generic analysis are bounding for Plant Hatch.

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

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

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: August 10, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.9.1, ''Refueling Equipment Interlocks,'' to provide alternative actions when the refueling equipment interlocks are inoperable.

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

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

The operation of refueling interlocks is explicitly assumed in the analyses of the control rod removal error and fuel loading error during refueling. Inadvertent criticality is prevented during the loading of fuel provided all control rods are fully inserted. The refueling interlocks accomplish this by preventing the loading of fuel into the core with any control rod withdrawn, or by preventing withdrawal of a rod from the core during fuel loading. Under existing TS when the refueling interlocks are inoperable, the current method of preventing fuel loading with control rods withdrawn is to prevent fuel movement. An alternate method to ensure that fuel is not loaded into a cell with a control rod withdrawn is to prevent control rods from being withdrawn and to verify that all control rods are fully inserted. The proposed TS Required Actions will require that a control rod block be placed in effect, thereby ensuring that control rods are not subsequently inappropriately withdrawn, and that all required control rods be verified to be fully inserted. This verification is in addition to the requirements to periodically verify control rod position by other TS requirements.

The proposed actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn, because a control rod withdrawal block is in place. Like the current TS requirements, the proposed actions will ensure that unacceptable operations are blocked. Hence, the proposed additional Required Actions provide an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn as does the current TS Required Action. Therefore, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The change in the TS requirements does not involve a change in plant design or to the analyzed condition of the reactor core during refueling. The proposed new Required Actions will ensure that control rods are not withdrawn and cannot be inappropriately withdrawn, because a block to control rod withdrawal is in place. Therefore, no new failure modes are introduced, and the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

As discussed in the Bases for the affected TS requirements, inadvertent criticality is prevented during the loading of fuel provided all control rods are fully inserted during the fuel insertion. The refueling interlocks function to support the refueling procedures by preventing control rod withdrawal during fuel movement and the inadvertent loading of fuel when a control rod is withdrawn. The proposed change will allow the refueling interlocks to be inoperable and fuel movement to continue only if a control rod withdrawal block is in effect and all control rods are verified to be fully inserted. These proposed Required Actions provide an equivalent level of protection as the refueling interlocks by preventing a configuration that could lead to an inadvertent criticality event. The refueling procedures will continue to be supported by the proposed Required Actions because control rods cannot be withdrawn and as a result, fuel cannot be inadvertently loaded when a control rod is withdrawn. Therefore, the proposed changes do not result in a significant reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: August 17, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 3.3.1.1–1, ''Reactor Protection System [RPS] Instrumentation,'' to remove one RPS function and modify another.

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.

Modifications to the Scram Discharge Instrument Volume (SDIV) System are being implemented to ensure that the SDIV high water level instrumentation will respond adequately to provide redundant, diverse trip functions for a Scram Discharge Volume (SDV) inleakage event. Since the scram function will be successfully performed, the removal of the low scram pilot air header pressure trip function does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The design criteria for the Scram Discharge System is contained in the Safety Evaluation Report on the BWR Scram Discharge System, which was transmitted by NRC letter dated December 9, 1980, to all BWR licensees. Modifications to the SDV System have been evaluated to demonstrate that the high water level instrumentation in the SDIV will respond adequately to provide the required trip function. No new system failure modes are created as a result of removing the low scram pilot air header trip, since the redundant and diverse SDIV high water level instruments will initiate a successful reactor scram. Therefore, the removal of the low scram pilot air header trip function does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The water level in the SDIV is monitored by both resistance-temperature type detectors and float switches. Redundancy and diversity in the instrumentation that initiates the scram signal is maintained even with the removal of the low scram pilot air header pressure trip function. Modifications to the SDIV System have been evaluated to demonstrate that the high water level instrumentation will respond adequately to provide the required trip function for an inleakage event. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of application for amendments: October 9, 2001 (TS 01–10).

Brief description of amendments: The proposed amendment would change the Sequoyah (SQN) Unit 2 Operating License Technical Specifications (TSs), specifically TS 6.8.4.h, ''Containment Leakage Rate Testing Program,'' to allow a one-time 5-year extension to the current 10-year test interval for the containment performance-based leakage rate test program.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), Tennessee Valley Authority (TVA), 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 extension to Type A testing does not increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident.

TVA performed an evaluation of the risk significance for the proposed increase to the Sequoyah Unit 2 Type A test frequency. The results of the TVA evaluation indicate that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in NRC Regulatory Guide (RG) 1.174, ''An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis.'' TVA's evaluation indicates that the increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population is non-risk significant (3.5E–7/reactor year and 7.72 person-rem, respectively).

The proposed test interval extension does not involve a significant increase in the consequences of an accident because research documented in NUREG–1493 determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C testing provides assurance that containment leakage from penetration leak paths (*i.e.,* valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests.

In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections periodically in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. Accordingly, TVA's proposed extension of the Type A test interval does not [significantly] increase the probability or consequences of an accident previously evaluated.

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

The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident since there are no physical changes made to the plant. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident.

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

The proposed change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG–1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal affect on risk because most potential leakage paths are detected by Type C testing.

Previous Type A leakage tests conducted on Sequoyah Unit 2 indicate that leakage from Unit 2 containment has been less than the 10 CFR 50 Appendix J leakage limit of 1.0 L_a. A review of previous Unit 2 Type A test results indicate at least a 10 percent margin exists below the 1.0 L_a leakage limit. These test results provide assurance that the proposed extension to the Type A test interval would not significantly reduce the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to pdr@nrc.gov.

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

Date of application for amendment: May 31, 2001, as supplemented September 14, 2001.

Brief description of amendment: The amendment revised the TMI–1 Technical Specifications (TSs) to incorporate Cycle 14 specific limits for the variable low reactor coolant system pressure-temperature core protection safety limits. These changes are reflected in revisions to Figures 2.1–1 and 2.1–3 of the TSs and the related Bases.

Date of issuance: October 23, 2001.

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

Amendment No.: 238.

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

*Date of initial notice in **Federal Register:*** July 11, 2001 (66 FR 36337).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 30, 2000, as supplemented by letter dated September 13, 2001.

Brief description of amendment: The amendment modified the Final Safety Analysis Report (FSAR) to reflect analysis of a HI-STORM 100 spent fuel cask system, spent fuel pool description and crane operations.

Date of issuance: October 26, 2001.

Effective date: October 26, 2001, and shall be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e).

Amendment No.: 174.

Facility Operating License No. NPF–21: The amendment revised the Final Safety Analysis Report.

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15918). The September 13, 2001, supplemental letter provided additional clarifying information, did not expand the scope of the original **Federal Register** notice, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 23, 2001, as supplemented by letter dated September 25, 2001.

Brief description of amendment: The amendment revised the Technical Specifications to eliminate the requirement to move control element assembly #43 for the remainder of Cycle 15.

Date of issuance: October 22, 2001.

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

Amendment No.: 235.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46478). The September 25, 2001, supplemental letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 2001.

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: July 18, 2001.

Brief description of amendment: The amendment changes Technical Specifications (TS) Definitions 1.12 and 1.25, the effect of which will be to allow either an allocated or a measured response time to be utilized for the sensors in the Reactor Protective System and Engineered Safety Features Actuation System instrument loops.

Date of issuance: October 29, 2001.

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

Amendment No.: 175.

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

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46479).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 24, 2000, as supplemented by letters dated June 18 and August 21, 2001.

Brief description of amendment: The amendment revises Technical Specification 3.8.3 regarding the lube oil inventories for the Grand Gulf Nuclear Station, Unit 1, Divisions I, II, and III emergency diesel generators (EDGs), and will result in additional margins for lube oil availability to provide for EDG operability for seven days following a postulated design basis accident.

Date of issuance: October 23, 2001.

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

Amendment No: 149.

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

*Date of initial notice in **Federal Register:*** January 24, 2001 (66 FR 7680).

The June 18 and August 21, 2001 supplemental letters did not change the scope of the original **Federal Register** notice or the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 1, 2001, as supplemented by letters dated October 9, 2001, and October 18, 2001. The supplemental letters provided clarifying information only and did not change the original proposed no significant hazards determination.

Brief description of amendments: The amendments revise Byron and Braidwood technical specifications (TS) surveillance requirement (SR) 3.7.2.1 and SR 3.7.2.2 to add a note stating that these surveillances are not required to be met until the first startup after September 27, 2001. This change is applicable to Byron Station Units 1 and 2, and Braidwood Unit 2 only. This change is not applicable to Braidwood Station, Unit 1, due to the recent restart of the unit after the refueling outage.

Date of issuance: November 1, 2001.

Effective date: November 1, 2001.

Amendment Nos.: 124, 124, 119, and 119.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 23, 2001 (66 FR 53643).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 1, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 2, 2000, as supplemented by letters dated September 4 and September 28, 2001.

Brief description of amendment: This amendment increases the spent fuel pool (SFP) storage capability, as a result of the SFP re-racking project, from the current capacity of 735 fuel assemblies to a new capacity of 1624 fuel assemblies. The amendment also approves additional temporary storage of up to 90 fuel assemblies in the fuel transfer pit to support a complete re-racking of the SFP. The increase in SFP storage capacity will provide a full core offload capability during the plant's Cycle 13 operation and enable the Davis-Besse facility to meet its storage needs through April 22, 2017, which is the expiration date for the current operating license.

Date of issuance: October 19, 2001.

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

Amendment No.: 247.

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

*Date of initial notice in **Federal Register:*** September 6, 2001 (66 FR 46656).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 4, 2001.

Brief description of amendments: This license amendment request: Deletes Technical Specification (TS) 1.7, Definitions-Reportable Event, and TS 6.6, Reportable Event—Action; Revise TS 6.5.3, Technical Review and Control—Activities, and TS Bases 4.0.3, Applicability.

Date of issuance: November 2, 2001.

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

Amendment No.: 248.

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

*Date of initial notice in **Federal Register:*** February 12, 2001 (66 FR 31708).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 22, 2001, as supplemented August 24, 2001.

Brief description of amendment: Revised Technical Specifications to allow the containment equipment door and airlock doors to be open during core alterations and fuel movement under administrative controls.

Date of Issuance: October 22, 2001.

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

Amendment No.: 120.

Facility Operating License No. NPF–16: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48287).

The August 24, 2001, supplement did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 24, 2000, as supplemented June 29, 2001.

Brief description of amendments: The amendments would approve changes to the updated final safety analysis report to incorporate a supplemental methodology into the analysis of steam generator overfill following a steam generator tube rupture.

Date of issuance: October 24, 2001.

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

Amendment Nos.: 256 and 239.

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

*Date of initial notice in **Federal Register:*** January 24, 2001 (66 FR 7682). The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 12, 2000, as supplemented by letters dated November 7, 2000, June 19, and August 17, 2001.

Brief description of amendments: The amendments revised the technical specifications to change the standard by which you test charcoal used in engineered safeguard features systems to American Society for Testing and Materials D3808–1989. These revisions are made in accordance with Generic Letter 99–02, ''Laboratory Testing of Nuclear-grade Activated Charcoal.''

Date of issuance: October 24, 2001.

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

Amendment Nos.: 257 and 240.

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

*Date of initial notice in **Federal Register:*** August 23, 2000 (65 FR 51356). The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 24, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 28, 2001, as supplemented by letters dated September 14, 18, and 27, 2001. The letters dated September 14, 18, and 27, 2001, provided clarifying information, and did not alter the NRC staff's conclusions regarding finding of no significant hazards consideration.

Brief description of amendment: The amendment evaluates the licensee's revised calculation methodology for assessment of consequences of design basis accidents, and revises Technical Specifications.

Date of issuance: October 23, 2001.

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

Amendment No.: 187.

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

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48288).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 23, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 12, 2001.

Brief description of amendment: The Amendment revises the Technical Specifications Bases Control Program to incorporate revisions to 10 CFR 50.59.

Date of issuance: October 25, 2001.

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

Amendment No.: 188.

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

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48289).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 25, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: April 12, 2001.

Brief description of amendment: The amendment revises the Technical Specifications surveillance test requirement SR 3.6.1.3.8, for excess flow check valves (EFCVs), to relax the 18-month EFCV surveillance frequency by limiting the number of tests to a "representative sample" every 18 months, such that each EFCV will be tested at least once every 10 years.

Date of issuance: October 26, 2001.

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

Amendment No.: 189.

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

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48289).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 2001.

No significant hazards consideration comments received: No.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: June 12, 2001.

Description of amendment request: The amendment changes Technical Specification (TS) 4.4.10 to incorporate alternative reactor coolant pump flywheel inspections and makes administrative wording changes to TSs 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3.

Date of issuance: October 22, 2001.

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

Amendment No.: 79.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38764).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 22, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 18, 2001, as supplemented October 10, 2001.

Brief description of amendment: The amendment (1) deletes a redundant requirement for valving out control rod drives, (2) revises control rod accumulator operability requirements, (3) adds the option to hydraulically isolate control rod drives, and (4) corrects an inconsistency describing when source range monitors are required to be operable during core monitoring.

Date of issuance: October 26, 2001.

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

Amendment No.: 123.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31711).

The supplement provided clarifying information to the application that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 26, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 15, 2001.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to (1) reflect the replacement of Monticello's licensed operator initial and requalification training programs with an accredited systems-approach-to-training program and (2) relocate the existing TS requirements for procedures, records, and reviews to the Operational Quality Assurance Plan.

Date of issuance: October 30, 2001.

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

Amendment No.: 124.

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

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48290).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 31, 2001.

Brief description of amendments: The amendments extend the implementation date for Amendment No. 184 for Unit 1 and Amendment No. 158 for Unit 2 from November 1, 2001, to November 1, 2003. Amendment Nos. 184 and 158 approved technical specification changes to incorporate requirements related to oscillation power range monitoring (OPRM) instrumentation. The implementation date extension is needed to provide additional time to address software deficiencies with the OPRM system identified in a June 29, 2001, General Electric report filed pursuant to part 21 of Title 10 of the Code of Federal Regulations.

Date of issuance: October 29, 2001.

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

Amendment Nos.: 196 and 172.

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

Date of initial notice in ***Federal Register:*** September 19, 2001 (66 FR 48291).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 30, 2000, as supplemented by letters dated May 7, June 13, 2001, and by internet memoranda dated June 28, July 3, July 23, and October 16, 2001.

Brief description of amendments: Amendment Application No. 217 is a request to revise the San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) operating license and technical specifications to remove certain requirements that have been determined to be unnecessary and modify requirements to provide flexibility during the decommissioning of SONGS 1. This change removes the need to perform activities that are not providing a benefit to safely maintain the spent fuel in the spent fuel pool. This change also provides some flexibility in the operation of the spent fuel pool during the decommissioning of SONGS 1.

Date of issuance: October 30, 2001.

Effective date: October 30, 2001, to be implemented within 30 days of issuance.

Amendment No.: Unit 1–160.

Facility Operating License No. DPR–13: The amendment revised the Operating License and the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 13, 2000 (65 FR 77924).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 30, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 6, 2001 (TS 01–05).

Brief description of amendments: The amendments revised the SQN Unit 1 and 2 Technical Specifications (TSs) by changing the surveillance requirements for verifying that containment isolation valves to be closed. More specifically, valves in high radiation areas may be verified by administrative means. In addition, valves which are locked sealed or otherwise secured do not need to be reverified closed and are eliminated from the scope of the surveillance.

Date of issuance: October 24, 2001.

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

Amendment Nos.: 271 and 260.

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44177). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 24, 2001.

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: April 23, 2001.

Brief description of amendment: The amendment updates the license by deleting obsolete information, correcting errors, and making administrative changes to enhance the context and provide consistency.

Date of Issuance: October 22, 2001.

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

Amendment No.: 206.

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

Date of initial notice in ***Federal Register:*** May 30, 2001 (66 FR 29363).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 22, 2001.

No significant hazards consideration comments received: No.

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

Dated at Rockville, Maryland, this 6th day of November 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–28399 Filed 11–13–01; 8:45 am]

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OFFICE OF MANAGEMENT AND BUDGET

Public Availability of Year 2001 Agency Inventories Under the Federal Activities Inventory Reform Act of 1998 (Public Law 105–270) (''FAIR Act'')

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Notice of public availability of agency inventories of activities that are not inherently governmental.

SUMMARY: Agency Inventories of Activities that are not Inherently Governmental are now available to the public from the agencies listed below, in accordance with the ''Federal Activities Inventory Reform Act of 1998'' (Public Law 105–270) (''FAIR Act''). This is the second release of the 2001 FAIR Act inventories. In addition, the Office of Federal Procurement Policy has prepared and has made available a summary FAIR Act User's Guide through its Internet site: *http://www.whitehouse.gov/OMB/procurement/index.html.* This User's Guide will help interested parties review 2001 FAIR Act inventories, and will also include the Website addresses to access agency inventories.

The FAIR Act requires that OMB publish an announcement of public