provisions of 10 CFR part 20, subpart K, General License for Storage of Spent Fuel at Power Reactor Sites, as specified in 10 CFR 72.212(b)(5), MYAPS is required to meet the physical protection requirements of 10 CFR 73.55 for an ISFSI at a reactor site. However, MYAPC has proposed to be exempted from the requirements of 10 CFR 72.212(b)(5) to ''protect the spent fuel against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth'' in 10 CFR 73.55 (with certain exceptions provided by 10 CFR 72.212(b)(5)). Instead, MYAPC has proposed alternative approaches to meet the provisions of portions of 10 CFR 73.55 related to the security organization, physical barriers, access requirements, detection aids, communications, and response requirements. The alternative measures for protection against radiological sabotage would meet the same high assurance objectives and the general performance requirements of 10 CFR 73.55 related to spent fuel storage at an ISFSI.

Environmental Impacts of the Proposed Action

The NRC has completed its evaluation of the proposed action and concludes that granting an exemption from the requirements of 10 CFR 72.212(b)(5) to protect the spent fuel against the design basis threat of radiological sabotage in accordance with the requirements of 10 CFR 75.55, thereby enabling MYAPC to implement alternative provisions of 10 CFR 73.55, would not have a significant impact on the environment.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (i.e., the ''no-action'' alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement related to Operation of Maine Yankee Atomic Power Station (July 1972).

Agencies and Persons Contacted

In accordance with its stated policy, on April 19, 2001, the staff consulted with Mr. Patrick Dostie of the State of Maine, Department of Human Services, Division of Health Engineering, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letters dated January 4, 2001, March 12, 2001, and April 4, 2001, which 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 ADAMS Public Library component on the NRC Web site, *http:www.nrc.gov* (the Electronic Reading Room).

Dated at Rockville, Maryland, this 5th day of June 2001.

For the Nuclear Regulatory Commission.

Michael K. Webb,

Project Manager, Section 1, Project Directorate IV & Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–14753 Filed 6–11–01; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from May 21, 2001 through June 1, 2001. The last biweekly notice was published on May 30, 2001 (66 FR 29349).

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 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 13, 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 Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: 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 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,. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: April 6, 2001.

Description of amendment request: The existing Oyster Creek Technical Specification (TS) Section 4.7.B.5 requires capacity testing of the Station Batteries and the Diesel Generator Starting Batteries at least once per 24 months during a plant shutdown. The proposed amendment request will allow the 24-month capacity test for the Diesel Generator Starting Batteries to be performed during plant shutdowns or during the 24-month on-line Diesel Generator inspection (TS 4.7.A.3). The proposed revision to Section 4.7.B.5.b also reflects this change in specified frequency.

Additionally, TS 4.7.A.5 is revised to delete the statement that the battery capacity test need not be performed if the installed batteries were replaced during the previous Diesel Generator on-line biennial inspection. This exception is no longer necessary because the battery capacity testing is not restricted to refueling outages based on the proposed change to Section 4.7.B.5.

TS 4.7.B.5.a is revised to delete the phrase ''* * * to be considered operable'' because all of the specified surveillances constitute operability criteria. The title of Section 4.7.B is revised to identify applicability to the Diesel Generating Starting Batteries. These additional proposed revisions are considered administrative changes, which clarify the existing TS.

TS 4.7 Bases is also revised to reflect the above specification changes. Section 4.7 Bases contained on page 4.7–3 are being relocated to Bases page 4.7–4. This relocation of the Bases is a purely administrative 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change to allow the batteries to be tested during the 24-month Diesel Generator inspection outage does not increase the probability of occurrence of an accident previously evaluated. No change is being made to equipment, equipment operation, or equipment requirements. If a Diesel Generator battery were to fail during the 24-month inspection, the availability of the Diesel Generator will not be affected because the Diesel Generator will already be out of service for the inspection. The change will allow the Diesel Generator out of service time during refueling outages to be reduced or eliminated, thereby reducing risk.

The change to allow the batteries to be tested during the 24-month Diesel Generator inspection outage does not increase the consequences of an accident previously evaluated. No change is being made to equipment, equipment operation, or equipment requirements. If a Diesel Generator battery were to fail during the 24-month inspection, the consequences of the battery failing are not affected.

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

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

The change to allow the batteries to be tested during the 24-month Diesel Generator inspection outage does not create the possibility of a new or different kind of accident from any previously evaluated. Moving the testing will not create a new possible failure type, it will only move the detection of a battery failure from the refueling outage to the 24-month Diesel Generator inspection outage.

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

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

The change to allow the batteries to be tested during the 24-month Diesel Generator inspection outage does not reduce a margin of safety. Since the Diesel Generator will already be out of service for the 24-month inspection, the margin of safety for the Diesel Generator 24-month inspection outage will not be affected. The change will allow the Diesel Generator out of service time during the refueling outage to be reduced or eliminated, thereby increasing the margin of safety during the refueling outage.

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

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

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

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: January 23, 2001.

Description of amendment request: The proposed amendment revises the requirements for containment integrity associated with the personnel and emergency air locks and other penetrations during fuel movement and refueling operations to allow these penetrations to remain open. One door in each of the emergency and personnel air locks must be capable of being closed and each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be capable of being closed by an isolation valve, blind flange, or manual valve. The supporting revised design basis fuel handling accident inside containment analysis will also incorporate alternative source term methodology in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.67 and Regulatory Guide 1.183, ''Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors,'' July 2000. Technical Specification (TS) 3.8.7 is also revised to provide equivalent isolation methods for other penetrations consistent with Babcock & Wilcox Owner's Group (BWOG) Standard Technical Specifications (STSs), Section 3.9.3.c.1, NUREG–1430, April 1995. TS 3.8.11 is added to specify the requirement to maintain at least 23 feet of water over the top of the reactor vessel flange and the actions required if this level is not maintained. TS Bases 3.8 is revised to provide a description of the plant conditions under which the personnel and emergency air locks and other penetrations including those consistent with the BWOG STSs, Section 3.9., may be open during fuel movement, and the administrative controls that would be in place. The surveillance requirements of TS 4.4.1.3 are also revised to identify the exception allowed by TS 3.8.6 under which both doors of the personnel and emergency air locks can be open.

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

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

Response: The proposed change would allow the personnel and emergency air lock doors and other penetrations to remain open during fuel loading and refueling operations. These penetrations were previously closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident inside containment (FHA). These penetrations are not initiators of any accident. The probability of a FHA is unaffected by the position of these penetrations.

The new FHA analysis utilizing an Alternative Source Term with an open containment demonstrates that the maximum doses are well within the acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183. In the event of a fuel handling accident, actual control room and offsite doses will be less than analyzed values because containment integrity will be restored following an evacuation of containment. As noted above, with the Alternative Source Term implementation, the acceptance criteria are also being revised. A direct comparison of the new Alternative Source Term dose consequences with the existing licensing basis FHA source term dose consequences is not practical due to the significant differences in methodology and assumptions.

However, a comparison of the previous thyroid and whole body dose results for the postulated TMI Unit 1 FHA Inside Containment documented in the TMI Unit 1UFSAR [updated final safety analysis report] Chapter 14 with the new dose results expressed in terms of Total Effective Dose Equivalent (TEDE), using the guidance in Regulatory Guide 1.183 Footnote 7, indicates that the new doses are not significantly higher than the previous dose results. The revised Alternative Source Term calculated doses remain well within the allowable acceptance criteria.

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

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

Response: The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design or method of operation of the plant beyond the standard functional capabilities of the equipment. The proposed change involves a change to the Technical Specifications that would allow the personnel and emergency air lock doors and other penetrations to be open during fuel loading and refueling operations within the containment. Having these doors and penetrations open does not create the possibility of a new accident. Administrative provisions will be made to ensure the capability to close the containment in the event of a FHA inside containment.

Therefore, the 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?

Response: This proposed change has the potential for an increased postulated accident dose due to a FHA Inside Containment; However, the analysis demonstrates that the resultant doses are well within the appropriate acceptance criteria. The margin of safety, as defined by 10 CFR 50.67 and Regulatory Guide 1.183, has been maintained. The offsite and control room doses due to a FHA with an open containment have been evaluated with conservative assumptions, which ensure the calculation bounds the postulated accident dose. Closing at least one door in each of the personnel and emergency air locks following the evacuation of the containment and closure of other open penetrations would reduce the control room and offsite doses in the event of a FHA inside containment and provides additional margin to the calculated doses.

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

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

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: January 29, 2001.

Description of amendment request: The proposed amendment revises the Technical Specifications (TSs) to remove the note from TS 4.5.4.1 that restricts the applicability of the specified engineered safeguards feature (ESF) systems leakage rate limit of 15 gallons per hour to the current operating Cycle 13 and establish this value as the permanent TS limit. This limit had previously been approved with the issuance of Amendment No. 215 on August 24, 1999, for Cycle 13 only. The proposed amendment also would implement a full scope alternative source term for Three Mile Island Nuclear Station, Unit 1, in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.67 and the guidance contained in Regulatory Guide 1.183.

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

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

The proposed permanent Technical Specification limit on ESF Systems leak rate is identical with the existing licensing basis value and is conservatively reevaluated for the limiting design basis Maximum Hypothetical Accident (MHA) using alternative source term methodology. Implementation of the alternative source term in accordance with Regulatory Guide 1.183 does not affect the design or operation of the facility, and therefore, does not significantly increase the probability of an accident previously evaluated. Based on the results of this reanalysis, it has been demonstrated that with the requested Technical Specification change, the offsite and control room dose consequences for this limiting event remain within the allowable dose criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.

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

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

The proposed permanent Technical Specification limit on ESF leak rate and implementation of the alternative source term in accordance with Regulatory Guide 1.183 does not affect the design, functional performance, or operation of the facility or of any equipment within the facility. Modifications supporting the proposed change have been evaluated and determined not to create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The proposed change involves implementation of the alternative source term in accordance with 10 CFR 50.67 and Regulatory Guide 1.183, and maintains the current Technical Specification limit on ESF Systems leak rate. The reanalysis of the limiting design basis MHA has been performed using conservative methodologies as specified in Regulatory Guide 1.183. Margin has been maintained to ensure that the accident analysis dose consequences bound the postulated event scenarios. The calculated offsite and control room dose consequences for this limiting event are within the acceptance criteria as specified in 10 CFR 50.67 and Regulatory Guide 1.183.

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

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

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Richard P. Correia, Acting.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: May 8, 2001.

Description of amendment request: The proposed amendment would revise the frequency of the Technical Specification (TS) surveillance requirement to check the movement of the control rods. Specifically, the frequency listed for this requirement in TS Table 4.1–3, ''Frequencies for Equipment Tests,'' would be changed from ''every 31 days'' to ''quarterly'' during reactor critical operations.

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability [...] or consequences of an accident previously evaluated.

This change to the frequency of performance of surveillance does not result in any hardware changes or nor does it change the response of control rods in performing their specified function. Therefore the change cannot affect the probability of occurrence of previously evaluated accidents.

The proposed frequency has been determined to be adequate to assure the reliability of reactor trip based on the conclusions in NUREG 1366 [''Improvements to Technical Specification Surveillance Requirements''] and the recommendations of GL [Generic Letter] 93–05 [''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation''].

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce a new failure mechanism or a new or different type of accident than those previously evaluated since there are no physical changes being made to the facility. Performance of the surveillance on the revised frequency will not have an adverse affect on the ability of the control rods to perform their intended function. The proposed change does not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in [a] margin of safety.

The proposed reduction in surveillance testing reduces the risk for causing dropped rods or reactor trips. This results in a slight improvement in the margin of safety by decreasing challenges to reactor components and safety systems.

The proposed surveillance frequency, as supported by the industry experience described in NUREG–1366, continues to provide the required assurance of control rod operability, such that safety margins established through the design and facility license, including the Technical Specifications, remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment is expected to result in a slight net improvement in [a] margin of safety. Hence the proposed change would not involve a significant reduction in [a] margin of safety.

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Richard P. Correia, Acting.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: May 10, 2001.

Description of amendment request: The proposed amendment would remove Technical Specification (TS) surveillance requirement (SR) 4.6.A.4 that requires each emergency diesel generator (EDG) to be given a thorough inspection at least annually following the manufacturer's recommendations. The requirement for the EDG inspection will be relocated to the Updated Final Safety Analysis Report and will be in accordance with the licensee controlled maintenance program. The inspection period required by the maintenance program will also be changed to specify that it will be ''in accordance with the manufacturer's recommendations.''

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability [...] or consequences of an accident previously evaluated.

There is no change to the design, function, or capability of the EDGs as a result of this change. Hence there is no change in the probability of occurrence of an accident previously evaluated.

The change does not affect the ability of the EDGs to mitigate the consequences of any accident previously evaluated; including the loss of coolant accident coupled with loss of offsite power. To the contrary, this change is structured to enhance the availability and reliability of the EDGs by tailoring the actual EDG maintenance program to the EDGs' operational history and experience. In addition, the surveillance testing requirements of TS Surveillance Requirements 4.6.A.1, 2 & 3 have not changed and are adequate to verify the operability of the EDG system. And, the Maintenance Rule Program at IP2 [Indian Point Unit 2] has established specific performance criteria for the EDGs. These performance criteria, and requirements to ensure the criteria are met, are not affected by this change.

The deletion of the surveillance requirement and controlling EDG maintenance using a licensee-controlled maintenance program does not alter or prevent the ability of the EDGs to perform their intended functions.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The EDG is not an accident initiator. The proposed change does not involve any physical design change or operational change. Thus a new failure mode is not introduced. In addition, the proposed change has been evaluated to not degrade the reliability of any existing system, structure, or component. Therefore, the proposed change does not create a new accident initiator or precursor, or create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in [a] margin of safety.

As a result of this change, there are no changes to IP2's design or to the IP2 TS safety limits, limiting safety system settings, or limiting conditions [for] operation. A single SR is replaced by a performance-based maintenance program.

The substitution of the performance-based maintenance program for the prescriptive SR is expected to increase the availability of the EDGs because the amount of time the EDGs are out-of-service for on-line maintenance will decrease. Reducing the number of plant operating hours that the unit is exposed to an out-of-service EDG improves rather than reduces the margin of safety. The substitution of the performance-based maintenance program for the prescriptive SR is expected to improve the reliability of the EDGs by minimizing the possibility of adverse results that may result from intrusive maintenance activities. The expected reliability improvement improves rather than reduces [a] margin of safety.

The transfer of control of EDG maintenance from the TS to a licensee-controlled EDG maintenance program is an administrative change. But the change is structured so that maintenance program changes must be evaluated using the 10 CFR 50.59 process. Use of the 10 CFR 50.59 process assures that future changes to the EDG maintenance program cannot significantly increase the likelihood of a malfunction of the EDGs. And use of the 10 CFR 50.59 process, instead of the license amendment process, allows Con Edison to optimize EDG maintenance in a timely manner to meet the intent of 10 CFR 50.65.

The proposed changes do not adversely affect the EDG's ability to function when required to mitigate any accident or licensing basis event. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in [a] margin of safety.

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

Attorney for licensee: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Richard P. Correia, Acting.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 11, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification definitions 1.12, ''Core Alteration;'' 3.9.1, ''Refueling Operations—Boron Concentration;'' 3.9.2, ''Refueling Operations—Instrumentation;'' and 3.9.11, ''Refueling Operations—Water Level—Reactor Vessel.'' The Bases for these Technical Specifications would also be modified to reflect the proposed changes to these definitions.

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 Technical Specification changes associated with the definition for Core Alteration and LCO [limiting condition for operation], applicability, action requirements and surveillance requirements of Sections 3.9.1, 3.9.2 and 3.9.11 will not cause an accident to occur and will not result in any change in operation of the associated accident mitigation equipment. The design basis accidents (fuel handling and boron dilution event) remain the same postulated events described in the Millstone Unit No. 2 Final Safety

Analysis Report (FSAR). Therefore, the proposed changes will not increase the probability of an accident previously evaluated.

The proposed LCO and Applicability changes are consistent with the design basis accident analyses of record. This will ensure that the accident mitigation equipment functions and associated equipment are available for accident mitigation as assumed in the associated accident analyses. The proposed surveillance requirement changes will continue to provide reasonable assurance of equipment operability. As a result, the accident assumptions and mitigation methods will not be adversely affected by the changes. Therefore, the proposed changes will not result in [an] increase in the consequences of accident[s] previously evaluated.

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

The proposed changes to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. There will be no adverse effect on plant operation or accident mitigation equipment. The response of the plant and the operators following an accident will not be different. In addition, 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.

The proposed LCO and Applicability changes are consistent with the design basis accident analyses of record. The proposed surveillance requirement changes will continue to provide assurance of equipment operability. The proposed changes do not involve any changes in the accident analyses, therefore, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 23, 2001.

Description of amendment request: The proposed amendment would remove the surveillance requirement to perform inspections of the Emergency Diesel Generators (EDGs) during shutdown conditions from Technical Specifications; although, inspections of the EDGs would continue to be performed in accordance with procedures prepared in conjunction with the recommendations of the manufacturer.

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 analysis which is based on the representations made by the licensee in the April 23, 2001 application, is presented below:

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

The Technical Specification change is associated with the surveillance requirement to perform inspections of the EDGs during shutdown conditions. The proposed change will remove this surveillance requirement from Technical Specifications; although, inspections of the EDGs will continue to be performed in accordance with procedures prepared in conjunction with the recommendations of the manufacturer.

Removal of the EDG inspection surveillance requirement from Technical Specifications does not verify operability or EDG functions assumed in the safety analysis. EDG inspections, which are maintenance activities that can be adequately controlled by plant procedures, will still be performed in accordance with the recommendations of the manufacturer. This will provide continued assurance the EDGs will be available when required.

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

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

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

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

The proposed change will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 2, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to not require the moderator temperature coefficient (MTC) determination in TS 4.1.1.4.2.c if the results of the MTC determinations required in TSs 4.1.1.4.2.a and 4.1.1.4.2.b are within a certain tolerance of the corresponding design values.

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

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

Under the proposed change, compliance with the TS[s] is maintained by measuring the beginning[-]of[-]cycle [(BOC)] temperature coefficients.

This change does not require a modification to any of the assumptions used in the input to the safety analyses. The assumptions were based on the current range of MTC allowed by TSs. The proposed change does not include a revision to the TS allowed range of MTC.

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

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

This change does not result in changing plant operation or any TS limits. The MTC will continue to be acceptably verified within specified limits. As described in the Combustion Engineering topical report, if the BOC MTC measurements are within the specified tolerance when compared to the design value, then the EOC [end-of-cycle] value is expected to fall within the design margin.

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

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

This change does not modify the range of allowed temperature coefficients. The surveillance program consisting of BOC measurements, of plant parameter monitoring, and of explicit EOC predictions will ensure that the MTC remains within the range of acceptable values.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 27, 2001.

Description of amendment request: The proposed amendment would revise TS 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' which requires the inspection of each reactor coolant pump (RCP) flywheel in general conformance with the recommendations of Regulatory Position C.4.b of NRC Regulatory Guide (RG) 1.14, Revision 1, ''Reactor Coolant Pump Flywheel Integrity,'' dated August 1975. The proposed change revises TS 5.5.7 to provide an exception to the recommendations of Regulatory Position C.4.b which would allow either a qualified in-place ultrasonic volumetric examination (UT) over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (i.e., magnetic particle testing (MT) and/or liquid penetrant testing (PT)) of exposed surfaces of the removed flywheel to be conducted at approximately 10-year intervals. The proposed change is in accordance with the NRC approved Improved Standard TS Generic Change Traveler TSTF–237, Revision 1.

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

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

An integral part of the Reactor Coolant System (RCS) in a Pressurized Water Reactor (PWR) is the Reactor Coolant Pump (RCP). The RCP ensures an adequate cooling flow rate by circulating large volumes of the primary coolant water at high temperature and pressure through the RCS. Following an assumed loss of power to the RCP motor, the flywheel, in conjunction with the impeller and motor assembly, provide sufficient rotational inertia to assure adequate core cooling flow during RCP coastdown.

Westinghouse Electric Corporation Topical Report WCAP–14535A, ''Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination,'' dated November 1996, provides the technical basis for the elimination of inspection requirements for RCP flywheels for all domestic Westinghouse plants. In the Safety Evaluation for WCAP–14535A, dated September 1996, the NRC stated that the evaluation methodology described in WCAP–14535A is appropriate and the criteria are in accordance with the design criteria of RG 1.14. RCP flywheel inspections have been performed for 20 years with no indications of service induced flaws. Flywheel integrity evaluations show a very high flaw tolerance for the RCP flywheels. Crack extension over a 60-year service life is negligible. Structural reliability studies have shown that eliminating inspections after 10 years of plant life will not significantly change the probability of failure.

The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed change does not alter or prevent the ability of structures, systems, and components (SSC) from performing their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Braidwood and Byron Stations' Updated Final Safety Analysis Report (UFSAR). The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Braidwood and Byron Stations' UFSAR.

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

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

The proposed change does not modify the design or function of the RCP flywheels. Based upon the results of WCAP–14535A, no new failure mechanisms will be introduced by the revised RCP Flywheel Inspection Program. As presented in WCAP–14535A, detailed stress analysis and risk assessments have been performed that indicate that there would be no change in the probability of failure for RCP flywheels if all inspections were eliminated. Flywheel integrity evaluations show that RCP flywheels exhibit a very high tolerance for the presence of flaws.

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

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

There is no significant mechanism for in-service degradation of the flywheels since they are isolated from the primary coolant environment. Additionally WCAP–14535A analyses have shown there is no significant deformation of the flywheels even at maximum overspeed conditions. Likewise, the results of RCP flywheel inspections performed throughout the industry and at the Braidwood Station and the Byron Station identified no indications that would affect flywheel integrity.

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

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

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 16, 2001.

Description of amendment request: The proposed amendments would change the reference in Technical Specification 5.5.6, ''Pre-Stressed Concrete Containment Tendon Surveillance Program,'' from Regulatory Guide 1.35, ''Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,'' Revision 3, 1989, to a reference to Subsection IWL, ''Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants,'' of Section XI, ''Inservice Inspection,'' of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and to delete the applicability of Surveillance Requirement (SR) 3.0.2 to TS Section 5.5.6. SR 3.0.2 allows the surveillance to be performed within 1.25 times the interval specified in the surveillance's frequency.

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 to Technical Specifications (TS) Section 5.5.6, ''Pre-Stressed Concrete Containment Tendon Surveillance Program,'' change the reference in TS Section 5.5.6 from Regulatory Guide (RG) 1.35, ''Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,'' Revision 3, 1989, to a reference to Subsection IWL, ''Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants,'' of Section XI, ''Inservice Inspection,'' of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, and to delete the applicability of Surveillance Requirement (SR) 3.0.2 to TS Section 5.5.6. SR 3.0.2 allows the surveillance to be performed within 1.25 times the interval specified in the surveillance's frequency. The proposed changes do not significantly effect the Tendon Surveillance Program, inspection frequencies, and acceptance criteria which provide the requirements for the performance of the primary containment tendon inspections at LaSalle County Station, Unit 1 and Unit 2.

The performance of a primary containment tendon inspection is not a precursor to any accident previously evaluated. Thus, the proposed changes to the performance of a primary containment tendon inspection do not have any effect on the probability of an accident previously evaluated.

The performance of primary containment tendon inspections does provide assurance that the primary containment will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, 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 an accident from any accident previously evaluated?

The proposed changes to TS Section 5.5.6, provide assurance that the primary containment will perform as designed and do not introduce any new modes of primary containment operation of failure mechanisms.

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

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

On August 8, 1996, the NRC published a final rule in the **Federal Register** (i.e., 61 **Federal Register** 41303) to amend 10 CFR 50.55a, ''Codes and standards,'' to incorporate by reference Subsection IWL of Section XI, of the ASME B&PV Code. Subsection IWL of Section XI, of the ASME

B&PV Code, provides rules for the inservice inspection and repair of the reinforced concrete and post tensioning systems of Class CC components. LaSalle County Station, Unit 1 and Unit 2, primary containments are Class CC components. The amended 10 CFR 50.55a required incorporation of Subsection IWL of Section XI, of the ASME B&PV Code, into inspection programs by September 9, 2001. We have developed an inspection program to implement Subsection IWL of Section XI, of the ASME B&PV Code. The proposed TS changes support this program.

The revised Tendon Surveillance Program, inspection frequencies, and acceptance criteria developed to implement Subsection IWL of Section XI, of the ASME B&PV Code, as required by 10 CFR 50.55a, provide acceptable requirements to perform inspections of the tendons in the LaSalle County Station, Unit 1 and Unit 2, primary containments. Thus, the proposed change to TS Section 5.5.6 will continue to ensure the integrity of the Unit 1 and Unit 2 primary containment tendons as required by the current TS.

Thus, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 19 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, PSEG Nuclear LLC, and Atlantic City Electric Company, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for Amendments: April 3, 2001.

Description of amendment request: The proposed amendment would revise the Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, technical specifications (TSs) in accordance with Technical Specification Task Force (TSTF) item TSTF–258, Revision 4. This TSTF has been previously reviewed and approved by the NRC as generically applicable to nuclear plants with improved standard TSs, such as PBAPS. The proposed amendment revises TS Section 5.0, ''Administrative Controls,'' to delete details of staffing requirements, eliminate specific details for working hour limits, clarify requirements for the Shift Technical Advisor position, add regulatory definitions for Senior Reactor Operators and Reactor Operators, revise the Radioactive Effluents Control Program to be consistent with the intent of Title 10 of the Code of Federal Regulations (10 CFR) Part 20, delete periodic reporting requirements for main stream relief valve openings, and revise radiological control requirements for radiation areas to be consistent with those specified in 10 CFR 20.1601(c).

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

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

The proposed TS changes are administrative in nature and do not impact the operation, physical configuration, or function of plant equipment or systems. The changes do not impact the initiators or assumptions of analyzed events, nor do they impact mitigation of accidents or transient events. Therefore, these proposed changes do not increase the probability of occurrence or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed TS changes are administrative in nature and do not alter plant configuration, require that new equipment be installed, alter assumptions made about accidents previously evaluated, or impact the operation or function of plant equipment. The proposed changes do not introduce any new modes of plant operation or make any changes to system setpoints. Therefore, these proposed changes do not create the possibility of a new or different kind of accident than previously evaluated.

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

The proposed TS changes are administrative in nature and do not involve physical changes to plant structures, systems, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions for operation, or design parameters for any SSC. The proposed changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis. Therefore, these changes do no involve any reduction in a margin of safety.

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

Attorney for Licensee: Mr. Edward Cullen, Vice President and General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 4, 2001.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) Section 1.7, Definitions—Reportable Events, and TS 6.6, Reportable Event Action, from the Davis-Besse Nuclear Power Station Operating License, and revise TS 6.5.3, Technical Review and Control—Activities, and TS Bases 4.0.3, Applicability. These changes are being proposed to delete TS requirements already required by Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), update the TS Bases to reflect recent changes made to 10 CFR 50.73, revise the approval authorizations for procedures, plant modifications, tests and experiments, and reflect recent changes made to 10 CFR 50.59.

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

The Davis-Besse Nuclear Power Station has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because no accident initiators, conditions or assumptions are affected by the proposed changes to delete Technical Specification (TS) 1.7, Definitions—Reportable Event, and TS 6.6, Reportable Event Action, from the Davis-Besse Nuclear Power Station (DBNPS) Operating License; and revise TS Bases 4.0.3, Applicability. Reportable Events are addressed by 10 CFR 50.73 and it is not necessary for the TS to include items already required by federal regulation. The proposed changes to TS Bases 4.0.3 would make these Bases consistent with the recent revision to 10 CFR 50.73. The proposed changes to the TS Index reflect the deletion of TS 1.7 and TS 6.6, Reportable Event Action, and are administrative changes.

The proposed changes to TS 6.5.3, Technical Review and Control—Activities, provide for the approval of activities affecting nuclear safety by personnel authorized by procedure. These changes continue to implement the DBNPS Quality Assurance

Program commitments. Qualification requirements for individuals performing reviews of activities affecting nuclear safety are not affected. Accordingly, there is no increase in the probability of an accident.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because no accident conditions or assumptions are affected by the proposed changes. The proposed changes do not alter the source term, containment isolation, or allowable releases. The proposed changes, therefore, will not increase the radiological consequences of a previously evaluated accident.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident initiators or assumptions are introduced by the proposed changes. The proposed changes do not alter any existing accident scenarios, or involve a modification or change in operation of any plant systems, structures, or components.

3. Not involve a significant reduction in a margin of safety because the proposed changes are administrative in-nature and do not reduce or adversely affect the capabilities of any plant structures, systems or components to perform their nuclear safety functions.

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

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

NRC Section Chief: Anthony J. Mendiola.

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 amendment requests: May 15, 2001.

Description of amendment requests: The proposed amendments would replace the current Technical Specification (TS) requirement to establish containment integrity within 8 hours if less than the specified minimum complement of A.C. or D.C. busses and equipment is operable in Modes 5 and 6. The proposed TS would require immediate suspension of operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies, and immediately initiate actions to restore the required busses and equipment to operable status, and to immediately declare the associated required residual heat removal loop(s) inoperable. The current Action requirement presents a scheduling and administrative burden during outages and extended shutdowns. In the addition, the proposed amendment would add options to the TS to allow containment penetration closure methods that are equivalent to those that are currently required during core alterations or movement of irradiated fuel in containment, and allow unisolation of some penetrations under administrative control. The additional options will allow flexibility in scheduling outage activities.

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

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

Probability of Occurrence of an Accident Previously Evaluated

The proposed changes to Action statements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 will eliminate current compensatory requirements that can only mitigate the consequences of accidents. The current requirements will be replaced with requirements that include measures to reduce the likelihood of accidents and assist in responding to malfunctions. The proposed requirements to immediately suspend operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies provide assurance that the applicable accidents, fuel handling and shutdown dilution accidents, will not occur by requiring cessation of activities that may cause them. The proposed requirements to immediately initiate actions to restore the required busses and equipment to operable status and to immediately declare associated required RHR loop(s) inoperable provide assurance operators can take timely corrective action for malfunctions that may lead to a dilution accident, and will take appropriate corrective actions for RHR malfunctions. Therefore, there is no adverse effect on accident initiators or precursors.

The proposed change to the Applicability requirements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 expands the conditions under which the T/S are invoked. The proposed change will assure that the electrical power is available for mitigation of a fuel handling accident, regardless of the operational mode of the plant. The proposed change only involves accident mitigation capabilities and does not affect any accident initiators or precursors.

The proposed changes to the LCO for T/S 3/4.9.4 will provide additional options for assuring closure of containment penetrations during core alterations or movement of irradiated fuel in containment. Containment closure provides only mitigation for the consequences of a fuel handling accident and does not affect the initiators or precursors of the accident.

The proposed change to the Surveillance requirements for T/S 3/4.9.4 allows the LCO to define the penetration status that is to be periodically verified. The effect of the proposed Surveillance change is bounded by the effect of the proposed LCO change as described above. Therefore, the proposed Surveillance change does not adversely affect any accident initiators or precursors.

Consequences of an Accident Previously Evaluated

The proposed changes to the Action requirements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 provide assurance that fuel handling and dilution accidents will not occur and that timely and appropriate responses can and will be taken for malfunctions, thereby reducing the likelihood that radioactive material will be released.

The proposed change to the Applicability requirements for Unit 1 T/S 3/4.8.2.2 and T/S 3/4.8.2.4 provides assurance that electrical power is available for mitigation of a fuel handling accident (FHA), regardless of the operational mode of the plant. Since the current Applicability requirement only provides this assurance in Modes 5 and 6, the proposed change will not increase the consequences of the accident.

The additional options provided by the proposed changes to the LCO for T/S 3/4.9.4 will mitigate the consequences of a fuel handling accident in containment as effectively as those specified by the current LCO. Additionally, the consequences of a FHA in containment determined by the accident analyses will not increase since the analyses do not credit mitigation by closure of containment penetrations.

The proposed change to the Surveillance requirements for T/S 3/4.9.4 only reflects the change proposed for the LCO. The effect of the proposed Surveillance change is bounded by the effect of the proposed LCO change as described above. Therefore, the proposed Surveillance change does not adversely affect the consequences of an accident.

The proposed changes to the Bases for the above identified T/S only provide explanatory information regarding the intent of the specifications and how they are to be implemented. The proposed Bases changes do not alter requirements of the associated T/S. Therefore, the effect of the Bases changes on accident initiators and precursors and on the consequences of an accident is bounded by the effect of the associated Action or LCO change as described above. The format changes do not alter any requirements.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

The proposed changes to Action statements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 to eliminate requirements to establish containment integrity does not affect existing, or create new, accident initiators or precursors because only existing passive accident mitigation features are involved. Implementation of the proposed new requirements to suspend operations involving core alterations, positive reactivity changes, and movement of irradiated fuel assemblies does not affect existing, or create new, accident initiators or precursors because these activities do not require the operation of existing equipment in a new or different manner, or involve the operation of new or different equipment. Implementation of the proposed new requirements to initiate actions to restore the required busses and equipment to operable status and to declare associated required RHR loop(s) inoperable does not affect or create new accident initiators or precursors because these activities are currently required by existing procedures and other T/S.

The proposed change to the Applicability requirements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 does not affect or create new accident initiators or precursors because it only expands the conditions under which the T/S are invoked.

The proposed changes to the LCO for T/S 3/4.9.4 to provide additional options for assuring closure of containment penetrations during core alterations or movement of irradiated fuel in containment does not affect or create new accident initiators or precursors because the changes involve only containment penetrations which are passive accident mitigation measures.

The proposed change to the Surveillance requirements for T/S 3/4.9.4 allows the LCO to define the penetration status that is to be periodically verified. The effect of the proposed Surveillance change is bounded by the effect of the proposed LCO change as described above. Therefore, the proposed Surveillance change does not affect or create new accident initiators or precursors.

The proposed changes to the Bases for the above identified T/S only provide explanatory information regarding the intent of the specifications and how they are to be implemented. The proposed Bases changes do not alter requirements of the associated T/S. Therefore, the effect of the Bases changes on accident initiators or precursors is bounded by the effect of the associated Action or LCO change as described above. The format changes do not alter any requirements.

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

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

The margins of safety pertinent to the proposed changes to Action statements for T/S 3/4.8.2.2 and T/S 3/4.8.2.4 are those associated with a FHA, a shutdown dilution event, and a RHR system malfunction. The applicable margin of safety for a FHA is that defined by the off site dose analyses for the accident. Since the analyses do not credit mitigation by the containment, the margin of safety is unaffected. The applicable margin of safety for a shutdown dilution event is the time available for operators to take action to preclude violating shutdown margin requirements. The proposed new Action requirements to immediately suspend operations involving positive reactivity changes, and to immediately initiate actions to restore the required electrical busses and equipment to operable status, would not decrease the margin of safety for a shutdown dilution event. The applicable margin of safety for a RHR system malfunction is the time available for operators to take action to restore decay heat removal capabilities. The proposed new actions requirements to immediately initiate actions to restore the required electrical busses and equipment to operable status and to immediately declare associated required RHR loop(s) inoperable would not decrease the margin of safety for a RHR system malfunction.

The margin of safety pertinent to the proposed changes to LCO for T/S 3/4.9.4 is that associated with a FHA. The applicable margin of safety for a FHA is that defined by the off site dose analyses for the accident. Since the analyses do not credit mitigation by the containment, the margin of safety is unaffected.

There is no margin of safety pertinent to the proposed changes to associated Applicability requirements, Surveillance requirements, and Bases for the above identified T/S. The format changes do not alter any requirements.

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

In summary, based upon the above evaluation, [Indiana Michigan Power Company (I&M)] has concluded that the proposed amendment involves no significant hazards consideration.

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

Attorney for licensee: David W. Jenkins, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: February 28, 2001.

Description of amendment request: The proposed amendment would change the Technical Specification (TS) to incorporate laboratory testing recommendations of Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' June 3, 1999.

The proposed charcoal testing changes and explicit reference to American Society for Testing and Materials (ASTM) D3803–1989 nuclear-grade activated charcoal test protocol do not affect engineered safety feature (ESF) ventilation system operation or performance, reliability, actuation setpoints, or accident mitigation capabilities. The proposed changes also do not affect the operation and performance of any other equipment important to safety at Cooper Nuclear Station (CNS).

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

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

The proposed charcoal testing changes and explicit reference to ASTM D3803–1989 nuclear-grade activated charcoal test protocol do not affect ESF ventilation system operation or performance, reliability, actuation setpoints, or accident mitigation capabilities. The proposed changes also do not affect the operation and performance of any other equipment important to safety at CNS. ASTM D3803–1989 is a more accurate and demanding test which ensures that the charcoal filter efficiencies assumed in the CNS accident dose analysis are maintained. The proposed changes involve ESF ventilation system charcoal testing only and do not affect accident initiators. Therefore the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the Updated Safety Analysis Report (USAR), as revised by the Design Basis Accident (DBA) radiological assessment calculational methodology revisions submitted to the U. S. Nuclear Regulatory Commission (NRC) under Reference 2.

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

The charcoal testing changes, and explicit reference to ASTM D3803–1989 nuclear-grade activated charcoal test protocol, do not affect ESF ventilation system operation or performance, or the operation and performance of any other equipment important to safety at CNS. The proposed changes clarify and explicitly identify the testing of the ESF ventilation system charcoal samples. No new or different accident scenarios, transient precursors, failure mechanisms, plant operating modes, or limiting single failures are introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident from that previously evaluated in the USAR, as revised by the DBA radiological assessment calculational methodology revision submitted to the NRC under Reference 2, is not created by this change.

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

The required performance of the ESF ventilation systems following a DBA is not impacted by utilizing a more demanding protocol for charcoal testing. Thus, the margin of safety assumed in the CNS accident analysis, as revised by the DBA radiological assessment calculational methodology revision submitted to the NRC under Reference 2, is maintained. Revising the TS to clarify charcoal testing methodology and explicitly referencing the charcoal absorber testing being performed does not affect ESF ventilation system performance or operation, or the operation and performance of any other equipment important to safety at CNS. Therefore, these 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: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 25, 2001.

Description of amendment request: The proposed amendment would change the Kewaunee Nuclear Power Plant Technical Specification 4.2 to remove the steam generator tube alternate repair criteria, because these alternate repair criteria, as approved, are not compatible with the replacement steam generators scheduled to be installed in the fall of 2001. In addition, the proposed amendment would make administrative changes revising the phrasing of text without altering technical content.

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.

Changing the technical specification within limits of the bounding accident analyses cannot change the probability of an accident previously evaluated or the currently licensed radiological consequence predicted by the analyses of record. Removal of an allowance for alternate repair criteria defaults to the more conservative repair criteria of plugging degraded tubes. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

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

Removal of alternate repair criteria from [Technical Specification] TS leaves in its place the more conservative, more restrictive criteria for plugging degraded steam generator tubes. Plugging degraded steam generator tubes is a currently licensed repair methodology for [Kewaunee Nuclear Power Plant] KNPP, is consistent with current plant design bases, and does not adversely affect any fission product barrier, nor does it alter the safety function of safety significant systems, structures and components or their roles in accident prevention or mitigation. Currently, licensed design basis accident and transient analyses of record bound the effect of plugging tubes. Thus, this proposal does not create the possibility of a new or different kind of accident.

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

The proposed change does not alter the manner in which Safety Limits, Limiting Safety System Setpoints, or Limiting Conditions for Operation are determined. It places TS 4.2 in a more conservative configuration than that previously approved for use by the [Nuclear Regulatory Commission] NRC. It conforms to plant design bases, is consistent with current safety analyses, and limits actual plant operation within analyzed and licensed boundaries. Removal of reference to use of alternate repair criteria from TS 4.2 and its Bases leaves existing and more conservative criteria in place. Thus, changes proposed by this request 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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: May 18, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications to delete a redundant requirement for valving out a control rod drive, revise control rod accumulator operability requirements, add the option to hydraulically isolate control rod drives, and correct an inconsistency in core monitoring describing when source range monitors are required to be operable.

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

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

Deleting the paragraph which specifies one specific pattern of control rod inoperability does not degrade the safe operation of the plant as inoperable control rods must still be analyzed to meet shutdown requirements.

Revising the operability requirements for control rod accumulators from "a nine-rod square array" to: "provided that no other control rod within two control rod cells in any direction has a:" is a clarification. No technical requirements are changed, therefore, the probability or consequences of previous evaluations of accidents have not been affected. This change will assure conformance with the Banked Position Withdrawal Sequence (BPWS) analysis documented in General Electric (GE) report NEDO–21231. No changes in plant equipment will occur.

The proposed change adds the option to hydraulically isolate the drive to prevent inadvertent drive withdrawal and not consider the accumulator inoperable. This provides a method of isolating a control rod drive with an inoperable accumulator in addition to electrical isolation when the control rod is fully inserted. A statement on when an inoperable accumulator is allowed is being relocated so that it also applies during refueling. Since in refueling, the plant is already shutdown, the accumulators are not required. As such, this change does not increase the probability or consequences of an accident previously evaluated.

A qualifier is being added that source range monitors (SRMs) only need to be functionally tested when there are more than two fuel assemblies present in any reactor quadrant. Criticality is not considered possible with two or less fuel bundles in each quadrant and adjacent to an SRM. Since this change will only allow bypassing SRM functional checks when two fuel bundles or less are present in each quadrant, this change cannot result in an inadvertent criticality. This proposed change would reduce surveillance testing to that time when the instrument is required to be operable and provide consistency between specifications.

The proposed Technical Specification changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. The proposed amendment does not introduce new failure modes. Based on the above justification, the proposed amendment will have no impact on the probability or consequences of an accident.

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

This change does not degrade the safe operation of the plant as inoperable control rods must still be analyzed to meet existing shutdown reactivity requirements. It will assure conformance with the Banked Position Withdrawal Sequence analysis documented in General Electric report NEDO–21231. No changes in plant equipment will occur.

Adding hydraulic isolation will not create the possibility of a new or different kind of accident from any accident previously analyzed.

Since this change will only allow bypassing SRM functional checks with two fuel bundles or less present in each quadrant, this change cannot result in an inadvertent criticality.

The proposed Technical Specification changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. The proposed amendment does not introduce new failure modes. Based on the above justification, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

Revising the control rod operability requirement does not degrade the safe operation of the plant.

Hydraulic isolation provides a method of isolating the drive in addition to the current electrical isolation. Both methods disarm the control rod drive and preclude the possibility of inadvertent drive withdrawal during subsequent operations. Adding applicability during refueling has little impact on safety as the drive is required to be fully inserted prior to isolation. As such, they do not involve a significant reduction in the margin of safety.

Since this change will only allow bypassing SRM functional checks with two fuel bundles or less present in each quadrant, this change cannot result in an inadvertent criticality.

Based on the above justification, the proposed Technical Specification change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Claudia M. Craig.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of amendment requests: May 4, 2001.

Description of amendment requests: The proposed amendments delete requirements from the Technical Specifications (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the technical specifications (TS) for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated May 4, 2001.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

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

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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: August 8, 2000.

Description of amendment request: The proposed change would add a new condition and associated required actions to Technical Specification (TS) 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs).'' The new condition and actions address the unique containment isolation features of the hydrogen-oxygen (H_2O_2) analyzer penetrations. The containment isolation barriers for the H_2O_2 analyzer penetrations consist of two PCIVs in series and a closed piping system outside primary containment. Editorial changes necessary to accommodate the addition of the proposed requirements were also proposed.

The licensee also requested approval for a proposed exception to the Susquehanna Steam Electric Station Final Safety Analysis Report commitments regarding conformance of the design of closed systems to the criteria of Section 6.2.4 of NUREG–75/087, Revision 1, 1975 (Standard Review Plan). The exception is related to the boundary valves between the H_2O_2 analyzer and the post-accident sampling system (PASS) and is necessary to permit the use of the H_2O_2 analyzer piping system outside primary containment as a redundant containment isolation barrier.

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

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

The proposed change adds a condition to LCO 3.6.1.3 to address the unique design of the H_2O_2 analyzer penetration. The H_2O_2 analyzer penetration isolation design requires that both PCIVs and the closed system be operable in order to support the single failure criteria and containment integrity. As part of the proposed change, an exemption [exception] to NUREG–75/087 guidance on closed systems for having all closed system boundary valves to be powered from a Class 1E power source is being requested. The proposed changes to Technical Specifications and Technical Specification Bases have no impact upon the safety functions of the H_2O_2 Analyzer PCIVs and closed system. The safety functions of these components are to maintain primary containment integrity by limiting leakage following an accident to within that assumed in the DBA [design-basis accident] LOCA [loss of coolant accident] Dose Analysis and to open to permit use of the H_2O_2 Analyzer systems post accident. The H_2O_2 Analyzer PCIVs and closed system will be maintained and leak rate tested in accordance with the Leakage Rate Test Program, thereby assuring that leakage from these components is maintained within the required limits. The design of these components is such that they meet the applicable design requirements with the exception of the PASS closed system boundary valves discussed above. However, the potential for a consequential failure of these valves has been evaluated and determined to be not credible. Thus, the proposed changes have no impact upon the H_2O_2 Analyzer PCIVs and closed system to perform their containment isolation function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As discussed above, the proposed change to the Technical Specifications does not impact upon the safety function of the H_2O_2 Analyzer PCIVs and closed system. The safety functions of these components are to maintain primary containment integrity by limiting leakage following an accident to within that assumed in the DBA LOCA Dose Analysis and to open to permit use of the H_2O_2 Analyzer systems post accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not affect the safety function of any plant system or component, and does not have any impact on plant operation. The proposed change does not involve a significant reduction in the margin of safety as currently defined in the bases of the applicable Technical Specification section. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Acting Section Chief: Richard Correia.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: May 21, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications to eliminate the response time testing requirements for the reactor protector system (RPS) signals of reactor high steam dome pressure and reactor vessel water level low.

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

Southern Nuclear Operating Company (SNC) has reviewed the proposed Technical Specifications changes described above and determined they do not involve a significant hazards consideration based on the following:

1. The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The purpose of the proposed changes is to eliminate response time testing requirements for select components in the RPS. However, because of the continued application of other existing Technical Specifications requirements, such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response time of the RPS will be maintained within the acceptance limits assumed in plant safety analyses. This will assure successful mitigation of an initiating event. The proposed Technical Specifications changes do not affect the capability of the associated systems to perform their intended function

within their required response time. The BWR Owners' Group (BWROG) has documented an evaluation in NEDO–32291, Supplement 1, ''System Analyses for the Elimination of Selected Response Time Testing Requirements'', which was submitted to the NRC for review and approval as a Topical Report in December 1997. The BWROG submitted additional information to the staff in Addendum 1 to NEDO 32291, Supplement 1 in November, 1998. Subsequently, the NRC approved the Topical Report by a Safety Evaluation Report (SER) issued in June, 1999.

This evaluation demonstrates that response time testing is redundant to the other Technical Specifications requirements listed in the proceeding paragraph. These other tests are sufficient to identify failure modes or degradation in instrument response time and ensure operation of the associated systems within acceptance limits. Furthermore, Addendum 1 to NEDO 32291, Supplement 1 clearly demonstrates defense-in-depth, such that from a realistic basis, there is no safety significance even if instrumentation loop response times are significantly longer than the loop bounding response times.

2. The proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed Technical Specifications changes do not affect the capability of the RPS to perform its intended function within the acceptance limits assumed in plant safety analyses. Periodic surveillance of these RPS instrument loop components will continue and may be used to detect degradation that could cause the response time characteristic to exceed the BRT [bounting response time] allowance.

3. The proposed changes do not involve a significant reduction in the margin of safety. The current Technical Specifications response times are based on the maximum allowable values assumed in the plant safety analyses, which conservatively establish the margin of safety. As described above, the proposed Technical Specifications changes do not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for the plant safety analyses. Plant and system responses to an initiating event will remain in compliance with the assumptions of the safety analyses; therefore, the margin of safety is not affected.

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

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

NRC Section Chief: Richard L. Emch, Jr.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: May 23, 2001.

Description of amendment request: The proposed amendment would change the Safety Limit Minimum Critical Power Ratios (SLMCPR) in Technical Specification (TS) 2.1.1.2 to reflect the results of a cycle-specific calculation.

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

The derivation of the revised SLMCPRs for Plant Hatch Unit 2 Cycle 17 for incorporation into the TS, and their use to determine cycle-specific thermal limits, have been performed using NRC-approved methods and procedures. The procedures incorporate cycle-specific parameters and reduced power distribution uncertainties in the determination of the value for SLMCPRs. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient. The basis of the MCPR Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling and the probability of fuel damage is not increased. Therefore, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes result only from a cycle-specific application of NRC-approved methods to the Unit 2 Cycle 17 core reload. These changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in the TS bases will remain the same. Cycle-specfic SLMCPRs are calculated using NRC-approved methods and procedures which are in accordance with the current fuel design and licensing criteria. The SLMCPRs remain high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a reduction in the margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: April 27, 2001.

Description of amendment request: The proposed amendments would revise Technical Specification 3.3.6, ''Containment Ventilation Isolation Instrumentation,'' to relax the slave relay test frequency from every 92 days to every 18 months.

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

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

The results of WCAP–13878 demonstrate that slave relays are highly reliable. WCAP–13878 also provides guidance to assure that slave relays remain highly reliable. The aging assessment concludes that the age/temperature-related degradation of all ND [normally deenergized] relays, and NE [normally energized] relays produced after 1992, is sufficiently slow such that a refueling frequency surveillance interval will not significantly increase the probability of slave relay failures. Finally, the evaluation of the auxiliary relays actuated during slave relay testing has concluded that based on the tests of the auxiliary relays performed during other equipment testing, reasonable assurance is provided that failures will be identified if the associated slave relays are tested on a refueling frequency.

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

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

The proposed changes do not alter the performance of the CVI [containment ventilation isolation] systems assumed in the plant safety analysis. Changing the interval for periodically verifying CVI slave relays (assuring equipment operability) will not create any new accident initiators or scenarios.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated for VEGP [Vogtle Electric Generating Plant].

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

The proposed changes do not affect the total CVI response assumed in the safety analysis since the reliability of the slave relays will not be significantly affected by the decreased surveillance frequency.

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

Based on the above safety evaluation, VEGP concludes that the changes proposed by this submittal satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards finding is justified.

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

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

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: December 20, 2000.

Description of amendment request: The proposed change will delete Condition 2.G, ''Reporting to the Commission,'' and Technical Specification 6.6.1.a, ''Reportable Event Action.''

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

Pursuant to 10 CFR 50.92, it has been determined that this request involves no significant hazards considerations. The determination of no significant hazards was made by applying the Nuclear Regulatory Commission established standards contained in 10 CFR 50.92. These standards assure that any changes to the operation of South Texas Project in accordance with this request consider the following:

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

Response: No.

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed changes. Therefore, this request does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This request involves administrative changes only. No actual plant equipment or accident analyses will be affected by the proposed change and no failure modes not bounded by previously evaluated accidents will be created. Therefore, this request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves administrative changes only.

No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety systems settings, or will not relax the bases for any limiting conditions of operation. Therefore, these proposed changes will not impact the margin of safety.

Conclusion: Based upon the analysis provided herein, the proposed amendments will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a reduction in a margin of safety. Therefore, the proposed amendments meet the requirements of 10 CFR 50.92 and do not involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 12, 2001.

Description of amendment request: The proposed amendment will revise Technical Specifications surveillance requirement 4.4.6.2.2.e,which refers to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, paragraph IWV–3427(b) as a requirement for demonstrating that each Reactor Coolant System Pressure Isolation Valve specified in TS Table 3.4–1 is operable. Part 10 of the ASME Operations and Maintenance (OM) Standards, OMa-1988, is currently the applicable code for these valves and does not have these requirements.

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

Pursuant to 10 CFR 50.91, this analysis provides a determination that the proposed change to the Technical Specifications described previously does not involve any significant hazards consideration as defined in 10 CFR 50.92.

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

This Technical Specification change only affects trending of valve leakage rate test results to anticipate the expected leakage rate performance of Reactor Coolant System pressure isolation valves. Redundant pressure isolation valves are included in the plant to ensure continued protection of lower pressure systems from exposure to the higher pressure of the Reactor Coolant System in the event that excessive leakage develops in an isolation valve. In addition, leakage rate tests of Reactor Coolant System pressure isolation valves will continue to be performed with no change in the accepted amount of leakage or frequency. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The limiting event associated with these valves is a Loss of Coolant Accident. This has already been reviewed as part of the South Texas Project Updated Final Safety Analysis Report. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

This proposed change only removes a requirement for trending of pressure isolation valve leakage rates. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no change in the design of the plant associated with this proposed license amendment. The only impact of this change is in the prediction of when a particular pressure isolation valve may have a leakage rate higher than what is allowed. Adverse test results will be addressed under the corrective action program and by application of the Maintenance Rule. Engineering analysis of test results can take into account special circumstances associated with a test that would affect the conclusions.

Leakage rate test measurements of South Texas Project Reactor Coolant System isolation valves will continue to be taken pursuant to the surveillance requirements of Technical Specification 4.4.6.2.2, which is consistent with the requirements of code OMa-1988, paragraph 4.2.2.3.e for analysis of leakage rates. Code OMa-1988, paragraph 6.3, requires records of tests, including analysis of deviations in test values. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 28, 2001.

Description of amendment request: The proposed amendment will revise the Technical Specifications (TS) to eliminate periodic response time testing requirements on selected sensors and selected protection channels, and will modify TS Section 1.0 Definitions for ''ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME'' and ''REACTOR TRIP SYSTEM (RTS) RESPONSE TIME'' to provide for verification of response time for selected components. Surveillances 4.3.1.2 and 4.3.2.2 will be modified consistent with the new definitions. The associated Bases will be revised.

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

Pursuant to 10 CFR 50.92, it has been determined that this request involves no significant hazards consideration. The determination of no significant hazards was made by applying the Nuclear Regulatory Commission established standards contained in 10 CFR 50.92. These standards assure that any changes to the operation of South Texas Project in accordance with this request consider the following:

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

Response: No.

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS [Reactor Trip System] and ESFAS [Engineered Safety Features Actuation System] instrumentation is being used; the time response allocations/modeling assumptions in the Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the SAR [Safety Analysis Report]. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This change does not alter the performance of the pressure and differential pressure transmitters and switches, Process Protection racks, Nuclear Instrumentation, and Logic Systems used in the plant protection systems. All sensors, Process Protection racks, Nuclear Instrumentation, and Logic Systems will still have response time verified by test before placing the equipment into operational service and after any maintenance that could affect the response time. Changing the method of periodically verifying instrument response times for certain equipment (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the equipment response time characteristics. Implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors and for Process Protection racks, Nuclear Instrumentation, and Logic Systems is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response time is within that assumed in the safety analysis. Based on the above, it is concluded that the proposed license amendment request does not result in a reduction in margin of safety.

Conclusion: Based on the preceding analysis, it is concluded that elimination of periodic equipment response time testing is acceptable and the proposed license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

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

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

NRC Section Chief: Robert A. Gramm.

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. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of application for amendment: March 1, 2001.

Brief description of amendment: The amendment increases the reactor core isolation cooling system surveillance test upper pressure limit.

Date of issuance: May 31, 2001.

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

Amendment No.: 139.

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

Date of initial notice in ***Federal Register:*** April 4, 2001 (66 FR 17964).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 6, 2000 (U–603329).

Brief description of amendment: The amendment relocates Technical Specification Figure 3.6.4.1–1, ''Secondary Containment Drawdown Time for 1500 cfm Boundary Leakage'' to plant procedures.

Date of issuance: June 1, 2001.

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

Amendment No.: 140.

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

Date of initial notice in ***Federal Register:*** November 29, 2000 (65 FR 71132).

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

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: December 21, 2000.

Brief description of amendment: The amendment revised Technical Specification (TS) 3.7.11 to allow plant operation to continue if the temperature of the Ultimate Heat Sink (UHS) exceeds the TS limit of 75 °F provided the water temperature, averaged over the previous 24-hour period, is at or below 75 °F. This operational flexibility only applies if the UHS temperature is between 75 °F and 77 °F. The action time requirements if the UHS temperature exceeds 77 °F, or if the 24-hour averaged value exceeds 75 °F still apply. An associated footnote that is no longer applicable was deleted, and the associated TS Bases were modified to reflect these changes.

Date of issuance: May 31, 2001.

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

Amendment No.: 257.

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

Date of initial notice in ***Federal Register:*** April 18, 2001 (66 FR 20007).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 31, 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: February 6, 2001, as supplemented by letter dated May 1, 2001.

Brief description of amendment: The amendment revised the Technical Specifications associated with the reactor coolant system leakage detection systems, to make them consistent with the requirements in NUREG–1432, ''Standard Technical Specifications, Combustion Engineering Plants.''

Date of issuance: May 29, 2001.

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

Amendment No.: 231.

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

Date of initial notice in ***Federal Register:*** March 7, 2001 (66 FR 13803).

The May 1, 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 May 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: January 24, 2000.

Brief description of amendment: Entergy Operations, Inc. requests revisions to the Grand Gulf Nuclear Station Technical Specifications which specify the minimum useable fuel oil inventories to be maintained in the Division 1, 2, and 3 Diesel Generator Fuel Oil Storage Tanks.

Date of issuance: May 24, 2001.

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

Amendment No: 147.

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

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15381).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 24, 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: November 7, 2000 as supplemented by letter dated March 23, 2001.

Brief description of amendments: The amendments would revise the technical specifications (TS) to extend the TS surveillance test interval (STI) from a 92-day STI to an 18-month STI, for the solid state protection system (SSPS) slave relay types that meet the acceptance criteria for the reliability assessments performed in accordance with the methodology described in the NRC approved Westinghouse Electric Corporation Topical Reports.

Date of issuance: May 31, 2001.

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

Amendment Nos.: 121, 121, 115, and 115.

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:*** February 21, 2001 (66 FR 11053).

The supplemental information 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 May 31, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 25, 2000.

Brief description of amendments: Revised the Action Statements associated with Technical Specification (TS) Table 3.3.7.5–1,''Accident Monitoring Instrumentation,'' concerning the Drywell Hydrogen/Oxygen (H_2/O_2) Concentration Analyzers, and the associated TS Bases.

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

Effective date: May 24, 2001.

Amendment Nos.: 151 and 115.

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

Date of initial notice in ***Federal Register:*** December 27, 2000 (65 FR 81929).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 24, 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: August 7, 2000, as supplemented February 6, 2001.

Brief description of amendment: This amendment will change Technical Specification (TS) Section Bases 3/4.3.1 and 3/4.3.2 to clarify the actions that must be performed when Steam and Feedwater Rupture Control System (SFRCS) components and SFRCS-actuated components are inoperable. Specifically, the changes will provide guidance on which TS actions are applicable for SFRCS-actuated components. The changes will also add a new TS 3/4.7.1.8 which would provide appropriate requirements for the Main Feedwater Control Valves and the Startup Feedwater Control Valves. Additionally, the changes add TS 3/4.7.1.9 which will provide requirements for the Turbine Stop Valves. The changes are consistent with the intent of NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants,'' Revision 1, April 1995.

Date of issuance: May 29, 2001.

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

Amendment No.: 246.

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

Date of initial notice in ***Federal Register:*** November 1, 2000 (65 FR 65342).

The supplemental information 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 May 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 6, 2001.

Brief description of amendment: The amendment revises the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs) Section 6.2, ''Organization,'' and Section 6.13, ''High Radiation Area'' to reflect the title change from Shift Supervisor to Shift Manager.

Date of issuance: June 1, 2001.

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

Amendment No.: 154.

Facility Operating License No. DPR–43: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 2, 2001 (66 FR 22031).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 30, 2000.

Brief description of amendments: The amendments approve the insertion of breakaway ceramic pins into the latches of eight double-leaf doors in the auxiliary building special ventilation zone in order to restrain the doors and reduce the frequency of open-door position alarms. The ceramic latch pins are designed to break at forces well below the differential pressure that would be generated in the auxiliary building as a result of a postulated high-energy line break (HELB), and thereby allow the doors to swing open and create a relief path from the auxiliary building. Therefore, the modification provides the restraints needed to reduce the frequency of open-door position alarms; but without impeding the doors' steam relief function that was assumed in the design-basis HELB analysis.

Date of issuance: May 30, 2001.

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

Amendment Nos.: 157 and 148.

Facility Operating License Nos. DPR–42 and DPR–60: Amendments revise the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** March 21, 2001 (66 FR 15928).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 16, 2001 (ULNRC–04390).

Brief description of amendment: The amendment revises Technical Specification 5.2.1.c to replace the title ''Vice President and Chief Nuclear Officer'' with ''Senior Vice President and Chief Nuclear Officer.''

Date of issuance: May 30, 2001.

Effective date: May 30, 2001, to be implemented within 60 days from the date of issuance.

Amendment No.: 145.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: April 4, 2001 (66 FR 17971)

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

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: December 12, 2000, as supplemented by letters dated January 8, and February 22, 2001.

Brief Description of amendments: The amendments revise Technical Specification Section 3.17 and associated Bases. The proposed changes will accommodate a vacuum-assisted fill technique for backfilling isolated reactor coolant system (RCS) loops from the active volume of the RCS.

Date of issuance: May 22, 2001.

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

Amendment Nos.: 226 and 226.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register***: March 21, 2001 (66 FR 15932).

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

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 5th day of June 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–14755 Filed 6–11–01; 8:45 am]

BILLING CODE 7590–01–U

NUCLEAR REGULATORY COMMISSION

Procedures for Providing Security Support for NRC Public Meetings/Hearings

The Nuclear Regulatory Commission (NRC) is revising its procedures for providing security support for all public NRC forums. This revision will provide a single set of procedures that will ensure consistency and uniformity in providing security support for these meetings. These procedures will be used by NRC headquarters and regional staff and are applicable to public hearings/meetings held at NRC headquarters buildings, other NRC space in the Washington, D.C. area, and/or regional locations to include space leased for the occasion. This **Federal Register** notice supersedes the previous **Federal Register** notice, entitled ''Security Support for NRC Meetings/Hearings,'' that was published on November 1, 1991 (56 FR 19451).

In order to balance the orderly conduct of government business with the right of free speech, the following procedures regarding attendance at NRC public meetings and hearings have been established:

Visitors (other than properly identified Congressional, press, and government personnel) may be subject to personnel screening, such as passing through metal detectors and inspecting visitors' briefcases, packages, etc.

Signs, banners, posters and displays will be prohibited from all NRC adjudicatory proceedings (Commission and Atomic Safety and Licensing Board Panel hearings) because they are disruptive to the conduct of the adjudicatory process. Signs, banners, posters and displays not larger than 18″×18″ will be permitted at all other NRC proceedings, but cannot be waved, held over one's head or generally moved about while in the meeting room. Signs, banners, posters and displays larger than 18″×18″ will not be permitted in the meeting room because they are disruptive both to the participants and the audience. Additionally, signs, banners, posters and displays affixed to any sticks, poles or other similar devices will not be permitted in the meeting room.

The presiding official will note, on the record, any disruptive behavior and warn the person to cease the behavior. If the person does not cease the behavior, the presiding official may call a brief recess to restore order and/or ask one of the security personnel on hand to remove the person.

For Further Information Contact: Calvin O. Byrd, Chief, Physical Security Branch, Division of Facilities and Security, Office of Administration, U. S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–7402.

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

For the Nuclear Regulatory Commission.

Thomas O. Martin,

Director, Division of Facilities and Security, Office of Administration.

[FR Doc. 01–14752 Filed 6–11–01; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Submission for OMB Review; Comment Request for Reclearance of a Revised Information Collection: SF 2802 and SF 2802B

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Public Law 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) has submitted to the Office of Management and Budget (OMB) a request for reclearance of a revised information collection. SF 2802, Application for Refund of Retirement Deductions (Civil Service Retirement System), is used to support the payment of monies from the Retirement Fund. It identifies the applicant for refund of retirement contributions. SF 2802B, Current/Former Spouse's Notification of Application for Refund of Retirement Deductions, is used to comply with the legal requirement that any spouse or former spouse of the applicant has been notified that the former employee is applying for a refund.

Approximately 32,100 SF 2802 forms are completed annually. We estimate it takes approximately 45 minutes to complete the form. The annual burden is 24,075 hours. Approximately 28,890 SF 2802B forms are processed annually. We estimate it takes approximately 15 minutes to complete this form. The annual burden is 7,223 hours. The total annual burden is 31,298 hours.

For copies of this proposal, contact Mary Beth Smith-Toomey on (202) 606–8358, or email to mbtoomey@opm.gov.

DATES: Comments on this proposal should be received on or before July 12, 2001.

ADDRESSES: Send or deliver comments to—

Ronald W. Melton, Chief, Operations Support Division, Retirement and Insurance Service, U.S. Office of Personnel Management, 1900 E Street, NW, Room 3349A, Washington, DC 20415–3540, and

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

For Information Regarding Administrative Coordination— Contact: Donna G. Lease, Team Leader, Budget and Administrative Services Division, Forms Analysis and Design, (202) 606–0623.

U.S. Office of Personnel Management.

Steven R. Cohen,

Acting Director.

[FR Doc. 01–14711 Filed 6–11–01; 8:45 am]

BILLING CODE 6325–50–P