Dated in Rockville, Maryland this 24th day of November, 1999.

For the Nuclear Regulatory Commission.

**John W. Craig,**

*NRC Standards Executive.*

[FR Doc. 99–31186 Filed 11–30–99; 8:45 am]

**BILLING CODE 7590–01–P**

## NUCLEAR REGULATORY COMMISSION

### Sunshine Act Meeting

**AGENCY HOLDING THE MEETING:** Nuclear Regulatory Commission.

**DATE:** Weeks of November 29, December 6, 13, and 20, 1999.

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

**STATUS:** Public and Closed.

**MATTERS TO BE CONSIDERED:**

*Week of November 29*

There are no meetings scheduled for the Week of November 29.

*Week of December 6—Tentative*

Wednesday, December 8

9:25 a.m. Affirmation Session (Public Meeting) (if needed)

*Week of December 13—Tentative*

Wednesday, December 15

9:25 a.m. Affirmation Session (Public Meeting) (if needed)

9:30 a.m. Meeting with Advisory Committee on Nuclear Waste (ACNW) (Public Meeting) (Contact: Dr. John Larkins, 301–415–7360)

Thursday, December 16

9 a.m. Meeting on NRC Response to Stakeholders' Concerns Location: (NRC Auditorium, Two White Flint North)

Friday, December 17

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

*Week of December 20—Tentative*

Wednesday, December 22

11:30 a.m. Affirmation Session (Public Meeting) (if needed)

* The schedule for Commission meeting is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

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

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, D.C. 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: November 26, 1999.

**William M. Hill, Jr.,**

*SECY, Tracking Officer, Office of the Secretary.*

[FR Doc. 99–31270 Filed 11–29–99; 10:49 am]

**BILLING CODE 7590–01–M**

## NUCLEAR REGULATORY COMMISSION

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

#### I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC) 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 November 6, 1999, through November 19, 1999. The last biweekly notice was published on November 17, 1999 (64 FR 62704).

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

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

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

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

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

By January 3, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

*Commonwealth Edison Company, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois*

*Date of amendment request:* October 12, 1999.

*Description of amendment request:* This proposed technical specification change removes the anticipatory reactor scram signal for turbine electro-hydraulic control (EHC) low oil pressure trip from the reactor protection system (RPS) trip function.

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

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

The proposed change removes the Turbine EHC Control Oil Pressure-Low scram function and the associated Limiting Safety System Setting (LSSS). The purpose of the Turbine EHC Control Oil Pressure scram is to anticipate the pressure transient which would be caused by imminent control valve closure on loss of control oil pressure. This function does not serve as an initiator for any accidents evaluated in Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). In addition, this trip function is not credited in any design basis event and is functionally redundant to the Turbine Control Valve Fast Closure RPS trip function during a postulated loss of EHC control oil event. The Turbine Control Valve Fast Closure will initiate a scram on a loss of control oil event coincident with turbine control valve closure.

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

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

The removal of this function does not represent a change in operating parameters or introduce a new mode of operation. The pressure switches associated with the Turbine Control Valve Fast Closure function provide equivalent protection from a loss of EHC oil event. For this reason, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Operation under the proposed amendment will not change any plant operation parameters, nor any protective system actuation setpoints other than removal of the Turbine EHC Control Oil Pressure-Low scram function. The scram function associated with the Turbine Control Valve Fast Closure provides equivalent protection for events involving fast turbine control valve closure including the loss of EHC control oil pressure. For this reason, eliminating the EHC Control Oil Pressure-Low scram function, which is redundant to other protective instrumentation, does not reduce the margin of safety.

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

*Attorney for licensee:* Ms. Pamela B. Stroebel, Senior Vice President and General Counsel, Commonwealth Edison Company, P.O. Box 767, Chicago, Illinois 60690–0767.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendment request:* September 23, 1999.

*Description of amendment request:* The proposed amendment would relocate items associated with instrumentation for toxic gas monitoring from the Technical Specifications (TSs) to the Updated Final Safety Analysis Report (UFSAR).

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

The proposed changes do not involve a significant hazards consideration because:

1. There is no significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature. The Specifications and associated Bases will be transferred verbatim to the UFSAR.

These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operating of the facility. The Limiting Safety Systems Settings and Safety Limits specified in the current TSs remain unchanged. Therefore, the proposed changes to the subject TS would not increase the probability or consequences of an accident previously evaluated.

2. The possibility of a new or different kind of accident from any accident previously evaluated has not been created.

As stated above, the proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility, and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently, no new failure modes are introduced as a result of the proposed changes, therefore, the proposed changes will not initiate any new or different kind of accident.

3. There has been no significant reduction in the margin of safety.

The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or physical design, the UFSAR design basis, accident assumptions are not affected. Therefore, the proposed changes will not result in a reduction in the margin of safety.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed changes do not represent a significant hazards consideration.

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

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

*NRC Section Chief:* Sheri Peterson.

*Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina*

*Date of amendment request:* November 3, 1999.

*Description of amendment request:* The amendments would revise Section 3.8.1, ''AC [alternating current] Sources—Operating,'' of the Technical Specifications. Specifically, this would revise: (1) Surveillance Requirement (SR) 3.8.1.9 to delete the power factor requirement from the diesel generator (DG) load rejection test; (2) SR 3.8.1.13 to allow performance of the diesel generator non-emergency automatic trip bypass test at any operational power level; and (3) SR 3.8.1.14 to allow performance of the 24-hour diesel generator run at any operational power level and delete the power factor requirement. No plant modification is involved with this proposed amendment.

*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, or

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

3. Involve a reduction in a margin of safety.

*First Standard*

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no effect on accident probabilities or consequences. The DGs and their associated emergency buses are not accident initiating equipment; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the equipment is not being modified by these proposed changes. In addition, the ability of the DGs to respond to a design basis accident will not be adversely impacted by these proposed changes. There will be no significant increased likelihood of causing a blackout of a safety bus by the proposed changes in testing. Therefore, there will be no significant impact on any accident consequences.

*Second Standard*

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

*Third Standard*

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. The equipment referenced in the revised TS for these proposed changes is already capable of performing as designed. No safety margins will be impacted.

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:* Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

*NRC Section Chief:* Richard L. Emch, Jr.

*Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina*

*Date of amendment request:* November 3, 1999.

*Description of amendment request:* The proposed amendments would revise Section 3.8.1, ''AC [alternating current] Sources—Operating,'' of the Technical Specifications. Specifically, this would revise: (1) Surveillance Requirement (SR) 3.8.1.9 to allow performance of the diesel generator (DG) load rejection test at any operational power level and to delete the power factor requirement; (2) SR 3.8.1.10 to allow performance of the diesel generator full load rejection test at any operational power level; and (3) SR 3.8.1.14 to allow performance of the 24-hour diesel generator run at any operational power level and delete the power factor requirement. No plant modification is involved with this proposed amendment.

*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, 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.

*First Standard*

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no effect on accident probabilities or consequences. The DGs and their associated emergency buses are not accident initiating equipment; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the equipment is not being modified by these proposed changes. In addition, the ability of the DGs to respond to a design basis accident will not be adversely impacted by these proposed changes. There will be no significant increased likelihood of causing a blackout of a safety bus by the proposed changes in testing. Therefore, there will be no significant impact on any accident consequences.

*Second Standard*

Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

*Third Standard*

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. The equipment referenced in the revised TS for these proposed changes is already capable of performing as designed. No safety margins will be impacted.

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:* Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

*NRC Section Chief:* Richard L. Emch, Jr.

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

*Date of amendment request:* October 7, 1999.

*Description of amendment request:* Grand Gulf Nuclear Station (GGNS) requests approval to revise its licensing basis for the release of fission products following an accident. The basis for the proposed change makes use of one of the insights established in NUREG–1465, ''Accident Source Terms for Light Water Nuclear Power Plants,'' which defines alternative source terms for use in the licensing of light water reactors. Specifically, this application credits the insight that there is a delay in the release of fission products from the reactor fuel following a postulated design basis loss-of-coolant accident (LOCA). The timing of fission product release from fuel perforation, i.e., gap activity release, is based on the boiling water reactor (BWR)—specific value of the timing of the gap activity release phase of a LOCA as calculated in the Boiling Water Reactor Owners Group (BWROG) Report, ''Prediction of the Onset of Fission Gas Release From Fuel in Generic BWR.'' This BWROG Report has been previously reviewed and approved by the Nuclear Regulatory Commission (NRC) staff. The licensing basis change to Updated Final Safety Analysis Report (UFSAR) Section 15.6.5.5.2 proposed by GGNS replaces the assumption of an instantaneous release of gap activity phase fission products into the drywell with a more accurate scenario in which the gap activity release is delayed by up to 121 seconds as calculated in the BWROG Report. Approval of this change will allow GGNS to increase the containment isolation valve closure times credited for limiting post-accident doses to both control room personnel and to offsite individuals. While this new basis would be applicable to all of the containment isolation valves, it addresses only the dose mitigation aspects of the closure requirements. There are currently some valves for which the closure time is limited based on other functional performance requirements (e.g., line break isolation). This submittal does not propose any changes that would eliminate any of these other requirements. The allowable closure times for these valves would not be affected by this proposed change.

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

GGNS staff has evaluated the proposed change to incorporate a delay in the post-accident fission product release into its licensing basis. This change recognizes one of the revised source term insights discussed in NUREG–1465. This change in the licensing basis will provide the basis for revising the Technical Requirements Manual to increase Primary Containment Isolation Valve (PCIV) maximum isolation times. These changes have been evaluated using the standards in 10CFR50.92 and it is concluded that they do not involve any significant hazards considerations. Specifically, the proposed change will not:

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

The proposed change takes credit for a new source term insight that recognizes that the fission product release from a fuel assembly is not instantaneous with a design basis accident. Implementation of this change into the licensing basis will be used to justify an increase in the maximum allowable PCIV isolation times. These changes do not affect the precursors for any accident or transient evaluated in Chapter 15 of the GGNS UFSAR. Therefore, there is no increase in the probability of any accident previously evaluated.

A plant specific radiological analysis has been performed to evaluate the effect on the dose consequences of extending the maximum allowable closure time. This evaluation considered the initial two-minute period of the accident during which, according to new source term insights developed in NUREG–1465 and in a BWROG report, fission product releases are not expected to occur. Releases from the break and from containment during this period consist of coolant radioactivity only. The total release during this period was found to result in an offsite dose of less than 0.60 rem. This dose represents only a small fraction of the LOCA dose evaluated in the UFSAR. As this submittal is for a limited scope application of the NUREG–1465 insights (in this case, timing and duration of the coolant activity phase) and addresses only the first 121 seconds of the accident scenario, the total long-term dose determined using the TID–14844 assumptions is not changed by this submittal.

In reality, the other insights offered in the NUREG would be expected to result in an overall dose reduction. In any event, the dose consequences of the proposed change do not result in an increase in the consequences of any accident previously evaluated.

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

The primary containment isolation system is designed to prevent, as much as practicable, the unfiltered release of radioactive material to the environs following an accident. As such, the system is relied upon for accident dose consequence mitigation. Neither the revision of the licensing basis to recognize that fission product releases are not instantaneous as is assumed in the current analysis, nor the extension of the valve closure times affects the ability of the valves to perform their accident mitigation function. It is also noted that the increased closure time allowables will only be applied to valves which do not have an alternate constraining performance requirement for closure time; the safety functions of other supported components and systems are not affected. Thus, the proposed change does not create the potential for a new or different kind of accident.

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

The proposed change revises the bases for the offsite dose calculation to credit, in the initial 2 minutes of the accident scenario, the fact that there is no fuel failure expected during this time. That is, for the first two minutes of the event, only coolant activity is released. The other assumptions, bases and methodologies for offsite dose calculations used to evaluate the long-term offsite dose consequences of accidents described in FSAR [Final Safety Analysis Report] Chapter 15 are not affected by this change. The margin between calculated dose consequences described in the FSAR and regulatory limits is not reduced.

A recent GGNS analysis of the LOCA scenario considering the only release in the first 121 seconds is from the reactor coolant resulted in an EAB [exclusion area boundary] dose of less than 1 rem thyroid during this period. The total dose for the 0- to 2-hour period is not expected to increase due to the delay in the fission product release; the total amount of radioactivity released will remain the same. Both the recently evaluated 2-minute dose and the 24.9 rem in two hours as presented in the UFSAR are insignificant in comparison to the 300 rem acceptance limit for this scenario. The GGNS SER [safety evaluation report] acknowledges the conservatism of the old analysis methodology. An independent analysis done by the staff during their evaluation of the GGNS FSAR estimated doses could decrease about 95% if the fission product release were to be delayed by 2 minutes.

The bases for PCIV closure times described in the Technical Specifications remain unchanged. The inconsistency between the assumption of immediate containment isolation in the dose analysis and allowable isolation valve closure times of one to two minutes is eliminated by this change. Plant specific analysis has shown that the expected dose resulting from the PCIVs remaining open during this period is insignificant.

Actual safety benefits are expected to result from valve performance and reliability improvements, elimination of unnecessary reports and system performance improvements such as minimization of water hammer events. Therefore, the increase in maximum isolation time for certain PCIVs proposed in this submittal will not result in a significant reduction in the margin of safety.

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

*Attorney for licensee:* Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* August 20, 1999.

*Description of amendment request:* The proposed license amendment would modify the Technical Specifications (TSs) to allow revision of the 4KV Engineered Safeguards Bus Undervoltage Relay Degraded Voltage calibration to be performed at an annual interval rather than its present refueling interval and change the bases to state that the degraded voltage relay setpoint tolerance is being changed from an ''as left'' reading to an ''as found'' reading. Additionally, the new calculations supporting the request identified a need to compensate for lack of voltage margin through reliance on manual action in lieu of full automatic voltage protection, as implied by Chapter 8 of the Updated Final Safety Analysis Report (UFSAR). Such actions would involve load manipulations following a loss of coolant accident (LOCA) with post LOCA conditions in combination with extremely low switchyard voltage. An additional limit of operation with a maximum of 5 Circulating Water pumps while in single 230KV auxiliary transformer operation is also added to the UFSAR.

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

1. The proposed changes to the degraded voltage relay setpoint tolerance and calibration interval are intended to reduce the total degraded voltage relay setpoint uncertainties. These changes will provide greater confidence that minimum voltages necessary to operate NSR [nuclear safety related] equipment are not exceeded. In combination, the proposed changes for degraded voltage relay setpoint tolerance and

calibration interval will reduce the probability that ES [engineered safeguards] buses will be separated from their offsite power source during low grid voltage conditions. This will reduce challenges to the onsite emergency power systems. The proposed changes will enhance the ability of the undervoltage protection scheme to perform in accordance with its intended design, and will improve the ability of the scheme to respond to low voltage conditions caused by malfunction of equipment important to safety.

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

2. The proposed setpoint tolerance and calibration interval changes are consistent with the specifications and intended design of the degraded voltage protection scheme and do not introduce the possibility of any new failure modes to the protection scheme or the electrical distribution system. The proposed changes reduce the probability of insufficient voltage to NSR loads and reduce the probability of separation of ES buses from the offsite power source. Therefore, operation of the facility in accordance with the proposed changes do not create a possibility of a new or different type of accident than any previously evaluated in the SAR.

3. The proposed setpoint tolerance and calibration interval changes are intended to reduce the total degraded voltage relay setpoint uncertainties. The changes will provide greater confidence that minimum voltages necessary to operate NSR equipment will not be exceeded. The proposed changes will also reduce the probability that the ES buses will be separated from their offsite power source during low grid voltage conditions. These effects will enhance the objective [of] providing a reliable source of power for BOP auxiliaries and [a] continuously available power supply for the ES equipment as required by TS [technical specification] 3.7 bases. Therefore, operation of the facility in accordance with the proposed changes 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:* Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

*NRC Section Chief:* Sheri R. Peterson.

*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:* November 3, 1999.

*Description of amendment requests:* The proposed amendments would allow use of fuel rods with ZIRLO cladding, specify an alternate methodology to determine the integral fuel burnable absorber (IFBA) requirements for Westinghouse fuel assemblies stored in the new fuel storage racks, and delete the designation of the fuel assembly types allowed in the spent fuel storage racks and the new fuel storage racks.

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

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

The proposed T/S [Technical Specification] change to allow storage and use of fuel rods clad with ZIRLO does not significantly increase the probability of occurrence of an accident. Fuel assemblies are not an initiator or precursor to any previously evaluated accident. The proposed T/S change does not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore, accident analysis results are not impacted. The operating limits are not changed and the analysis methods to demonstrate operation within the limits remain in accordance with NRC-approved methodologies. Other than the changes to the fuel rod cladding there are no physical changes to the plant associated with this T/S change. A safety analysis is still required to be performed for each specific reload cycle to demonstrate compliance with fuel safety design bases. The 10 CFR 50.46 emergency core cooling system acceptance criteria are applied to the ZIRLO clad fuel rods. The use of fuel assemblies containing ZIRLO clad fuel rods does not result in a change to the reload design and safety analysis limits. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material used, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks does not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. This alternate methodology is more conservative with respect to determining the reactivity of the stored fuel assemblies than the methodology currently specified in the T/S. Therefore, the probability of an accidental criticality is less with the proposed T/S change than currently assumed. Since a criticality accident is precluded by the proposed T/S change, the consequences of a criticality accident are not changed by the use of this alternate methodology.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is administrative, and does not alter the design and analysis requirements that ensure storage of fuel in safe configurations. The existing T/S requirements for maximum enrichment, reactivity, and spacing of fuel assemblies in the spent fuel storage racks and new fuel storage racks are not altered by this change.

Based on the above discussions, design basis accident analyses affected by these T/S changes remain valid, and the consequences of an accident previously evaluated are not significantly increased by these changes.

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

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

The proposed T/S change to allow storage and use of fuel rods clad with ZIRLO cannot create a new or different kind of accident. Fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for fuel assemblies with Zircaloy-4 clad fuel rods. The design and performance criteria continue to be met and no new failure mechanisms have been identified. Since the original design criteria are met, the ZIRLO clad fuel rods cannot be an initiator for any new accident. The ZIRLO cladding material offers improved corrosion resistance and structural integrity. The proposed changes do not affect the design or operation of any other system or component in the plant. The safety functions of the other structures, systems, or components are not changed in any manner, nor is the reliability of any other structure, system, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any other facility design feature, structure, or system. No new or different types of permanent plant equipment are installed by this proposed T/S change. In addition, the use of ZIRLO fuel assemblies does not involve any alterations to permanent plant equipment or plant operating procedures that would introduce any new or unique operational mode or accident precursor.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks ensures that a conservative methodology is used to verify the licensing basis reactivity limits are not exceeded. The proposed change does not affect any permanent plant equipment or plant operating procedures, and cannot be an initiator of an event.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change only. The proposed change does not affect any permanent plant equipment or plant operating procedures, and cannot be an initiator of an event.

Since there is no change to the permanent facility or plant operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, it is concluded that the change does 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 proposed T/S change to allow storage and use of fuel rods clad with ZIRLO does not change the reactor fuel reload design and safety analysis limits. The use of these fuel assemblies takes into consideration the core operating conditions allowed in the T/S. For each cycle reload core, the fuel assembly design and core configuration are evaluated using NRC-approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects. The design basis and modeling techniques for fuel assemblies with Zircaloy-4 clad fuel rods remain valid for fuel assemblies with ZIRLO clad fuel rods. Use of ZIRLO cladding material has no effect on the criticality analysis for the spent fuel storage racks and the new fuel storage racks. Furthermore, it has no effect on the thermal-hydraulic and structural analysis for the spent fuel pool. Therefore, the design and safety analysis limits specified in the T/S are maintained with this proposed change.

The proposed T/S change to specify an alternate NRC-approved methodology used to determine the IFBA requirements for Westinghouse fuel assemblies stored in the new fuel storage racks ensures that a conservative methodology is used to verify the licensing basis reactivity limits are not exceeded. Therefore, the existing T/S margin for reactivity control in the new fuel storage racks is maintained by this proposed change.

The proposed T/S change to delete designation of the fuel assembly types allowed in the spent fuel storage racks and new fuel storage racks is an administrative change, and does not alter any of the existing T/S limits governing storage and use of reactor fuel.

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

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

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

*Date of amendment request:* October 16, 1998, as supplemented by letters dated December 30, 1998, May 10, June 15, July 30, August 2, 11, 16, 19, 27, September 10, and 30, 1999.

*Description of amendment request:* Associated with a Niagara Mohawk Power Corporation (NMPC or the licensee) application to convert from the Curent Technical Specifications (CTS) for the Nine Mile Point Nuclear Power Station, Unit No. 2, to Improved Technical Specifications (ITS) as contained in Revision 1 of NUREG–1433, and Revision I of NUREG–1434, ''Standard Technical Specifications for General Electric Plants, BWR/4 and BWR/6'' dated April 1995, the licensee proposed to allow two hydrogen recombiners to be inoperable for up to 7 days provided that the alternate hydrogen control system is found to be acceptable to the NRC staff as described below.

CTS 3.6.6.1 ACTION only permits one hydrogen recombiner to be inoperable. If two hydrogen recombiners are inoperable, CTS 3.0.3 is entered. CTS 3.6.6.1 ACTION has been modified to incorporate Standard Technical Specification (STS) 3.6.3.1 ACTION B which allows two hydrogen recombiners to be inoperable for up to 7 days. The use of STS 3.6.3.1 ACTION B is allowed, as specified in a Bases Reviewer's Note, provided that the alternate hydrogen control system is found to be acceptable to the NRC staff. Therefore, the licensee proposed to allow credit be taken for an alternate hydrogen control system in the event of both hydrogen recombiners are determined to be inoperable for up to 7 days.

*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:

In accordance with the criteria set forth in 10 CFR 50.92, NMPC has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change extends the functional test frequency of the hydrogen recombiner system. The hydrogen recombiners are not considered as initiators for any previously evaluated accidents. Therefore, the probability of an accident previously evaluated is not significantly increased. The proposed change does not impact the Surveillance Requirement itself nor the way in which the Surveillance is performed. The proposed change does not affect the availability of the hydrogen recombiners to mitigate an accident because of the availability of the redundant hydrogen recombiner. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not involve a significant increase in the 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 change does not involve any design changes, plant modifications, or changes in plant operation. The system will continue to function in the same way as before the change. In addition, the Surveillance Requirement itself and the way the Surveillance is performed will remain unchanged. Furthermore, a historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The design, function, and OPERABILITY requirements for the hydrogen recombiner system are unchanged with this proposed revision. Although the proposed change will result in an increase in the interval between surveillance tests, the impact on hydrogen recombiner availability is small based on the redundant hydrogen recombiner, and there is no evidence of any failures that would impact the availability of the hydrogen recombiners. Therefore, the assumptions in the licensing basis are not impacted, and 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

*Attorney for licensee:* Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

*NRC Section Chief:* Sheri R. Peterson.

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

*Date of amendment request:* October 25, 1999.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications (TSs) to add the Oscillation Power Range Monitor (OPRM) Upscale function and allow the proposed activation of the OPRM function of automatically detecting and suppressing reactor instability conditions. Activation of the OPRM is in response to Generic Letter 94–02, ''Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors,'' licensee's associated commitment to implement stability solution Option III as described in Licensing Topical Report NEDO–31960–A, ''BWR Owners' Group Long-Term Stability Solutions Licensing Methodology,'' and previous Nine Mile Point Unit 2 (NMP2) License Amendment 80 dated March 31, 1998. The proposed changes would add the OPRM as a Reactor Protection Sytem (RPS) Functional Unit, including operability requirements and surveillance tests. Specifically, the proposed amendment would revise TS 2.2, ''Limiting Safety System Settings,'' TS 3/4.3.1, ''Reactor Protection System Instrumentation,'' TS 3/4.4.1, ''Recirculation System,'' and TS 6.9.1.9, ''Administrative Controls-Core Operating Limits Report.'' The proposed changes to support activation of the OPRM function are generally consistent with the changes proposed in Licensing Topical Report NEDC–32410P–A, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Plus Option III Stability Trip Function,'' Supplement 1, dated November 1997. The licensee's submittal also provides changes to the associated TS Bases and the TS Index (page ix).

The proposed changes would be made to NMP2's current TS, as well as to NMP2's improved TS addressed in a previous notice (64 FR 56518, October 20, 1999).

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

1. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of the OPRM Upscale functional unit to TSs involves a system that is intended to detect the symptoms of instability events and initiate mitigative actions. The worst case failure of the system involved would be a failure to initiate mitigative actions (i.e., scram), but no failure can cause an accident. The removal of certain RCS [Recirculation System] operational restrictions is justified with the addition of the OPRM functional unit which will provide an automatic scram in the event of reactor instabilities. Therefore, the proposed change will not result in a significant increase in the probability of any accidents previously evaluated.

The addition of the OPRM Upscale functional unit to the NMP2 TSs will permit activation of the OPRM. Activation of the OPRM, together with the NUMAC-PRNM, provides NMP2 the ability to detect and suppress reactor instabilities. The existing RPS functional units as well as other plant equipment will continue to perform their intended function in the event of an accident. The addition of the OPRM functional unit fulfills the intended purpose of the TS-required RCS operational restrictions. Therefore, the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of the OPRM Upscale functional unit to the NMP2 TSs will permit activation of the OPRM. Activation of the OPRM, together with the NUMAC–PRNM, provides NMP2 the ability to detect and suppress reactor instabilities. The OPRM is a mitigative system whose addition as an RPS functional unit will not create the possibility of a new or different accident or adversely affect existing RPS functional units. The worst case failure of the systems involved would be failure to initiate mitigative actions, but no failure can cause an accident. Except for the activation of the OPRM, no new plant configurations are created. The OPRM Upscale functional unit fulfills the intended purpose of the existing TS-required RCS operational restrictions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The proposed TS changes will not adversely affect the performance characteristics of RPS instrumentation nor will it affect the ability of the subject instrumentation to perform its intended function.

The addition of the OPRM Upscale functional unit to the NMP2 TSs will permit activation of the OPRM. Activation of the OPRM, together with the NUMAC-PRNM, provides NMP2 the ability to detect and suppress reactor instabilities (stability solution Option III) thereby meeting the requirements of GDC [General Design Criteria] 10 and 12. The NRC has reviewed and accepted the Option III methodology described in Licensing Topical Report NEDO-31960–A and concluded that the solution will provide the intended function. The surveillance testing and frequencies proposed will assure reliability of the OPRM Upscale function. The purpose of the existing TS operational restrictions on the RCS will be met by the automatic scram feature of the OPRM.

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

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

*Attorney for licensee:* Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

*NRC Section Chief:* Sheri Peterson.

*PECO Energy Company, Docket Nos. 50–352 and 50–353, Limerick Generating Station, (LGS) Units 1 and 2, Montgomery County, Pennsylvania*

*Date of amendment request:* October 14, 1999.

*Description of amendment request:* The proposed amendments, if approved, would revise the LGS, Units 1 and 2, Technical Specifications (TSs), Sections 2.2., ''Safety Limits and Limiting Safety System Settings,'' and 3.0/4.0, ''Limiting Conditions for Operation and Surveillance Requirements.'' The proposed revisions are required to support installation of a new Power Range Neutron Monitoring (PRNM) System and incorporate long-term thermal-hydraulic stability solution hardware.

*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.

As discussed in the Nuclear Measurement Analysis & Control (NUMAC) PRNM [Power Range Neutron Monitor] Licensing Topical Report (LTR), the NUMAC PRNM modification and associated changes to the TS involve equipment that is designed to detect the symptoms of certain events or accidents and initiate mitigating actions. The worst case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram or rod block), but no failure can cause an accident. The PRNM replacement system is designed to perform the same operations as the existing Power Range Monitor System and meets or exceeds all operational requirements. Therefore, it is concluded that the probability of an accident previously evaluated is not increased as a result of replacing the existing equipment with the PRNM equipment.

The PRNM System reduces the need for tedious operator actions during normal conditions and allows the operator to focus more on overall plant conditions. The automatic self-test and increased operator information provided with the replacement system are likely to reduce the burden during off-normal conditions as well. The replacement equipment qualifications fully envelope the environmental conditions, including electromagnetic interference, in the LGS control room.

The replacement equipment has been specifically designed to assure that it fully meets the response time requirements in the worst case. As a result, due to statistical variations resulting from the sampling and update cycles, the response time is typically faster than required in order to assure that the required response time is always met. Setpoints are changed only when justified by the improved equipment performance specifications and by setpoint calculations which show that safety margins are maintained. There is no impact to the Control Rod Drop accident analysis because the PRNM System maintains all existing system functions with a reliability equal to or better than the existing Power Range Monitor System.

The replacement equipment includes up to 5 LPRM [Local Power Range Monitor] inputs on a single module compared to one per module on the current system. Up to 17 LPRM signals are processed through one preprocessor. The recirculation flow signals are processed in the same hardware as the LPRM processing. The net effect of these architectural aspects is that there are some single failures that can cause a greater loss of ''sub-functionality'' than in the current system. Other architectural and functional aspects, however, have an offsetting effect. Redundant power supplies are used so that a single failure of Reactor Protection System (RPS) AC power has no effect on the overall PRNM System functions while still resulting in a half scram as does the current system. Continuous automatic self-test also assures that if a single failure does occur, it is much more likely to be detected immediately. The net effect is that from a total system level, unavailability of the safety-related functions in the replacement system is equal to or better than the current Power Range Monitor System.

Based on the extensive and thorough verification and validation program used in the PRNM design and field operating experience, common cause failures in software controlled functions are judged to not be a significant failure mode.

However, in spite of that conclusion, means are provided within the system to mitigate the effects of such a failure and alert the operator. Therefore, such a failure, even if it occurred, will not increase the consequences of a previously evaluated accident.

To reduce the likelihood of common cause failure of software controlled functions, thorough and careful verification and validation activities are performed both for the requirements and the implementing software design. In addition, the software is designed to limit the loading that external systems or equipment can place on the system, thus significantly reducing the risk that some abnormal dynamic condition external to the system can cause system functional performance problems due to processing ''overload'' (i.e., ''slowing down'' or stopping the processing).

As a conservatism, however, despite these verification and validation activities, common cause failures of software-controlled functions due to residual software design faults are assumed to occur. Both the software and hardware are designed to manage the consequences of such failure (and also cover potential common cause hardware failures). Safety outputs are designed to be fail safe by requiring dynamic update of output modules or data signals, where failure to update the information is detected by simple receiving hardware, which, in turn, forces a trip. This aspect covers all but rather complex failures where the software or hardware executes a portion of the overall logic but fails to process some portion of new information (inputs ''freeze'') or some portion of the logic (outputs ''freeze'').

To help reduce the likelihood of complex failures, a watchdog timer is used which is updated by a very simple software routine that in turn monitors the operational cycle time of all tasks in the system. The software design is such that as long as all tasks are updated at the design rate, it is likely that software controlled functions are executing as intended. Conversely, if any task fails to update at the design rate, that is a strong indication of at least some unanticipated condition. If such a condition occurs, the watchdog timer will not be updated, the computer will be automatically restarted, and the system will detect an abnormal condition and provide an alarm and trip.

The information available to the operator is at least the same as with the current system and, in many cases, improved. No actions are required by the operator to obtain information normally used and equivalent to that available with the current equipment. However, the replacement system does provide more directly accessible information regarding the condition of the equipment, including automatic self-test, which can aid the operator in diagnosing unusual situations beyond those defined in the licensing basis.

In summary, the reliability of the new PRNM System and its ability to detect and mitigate abnormal flux transients have either remained the same or improved over the existing Power Range Monitor System. Since these postulated reactivity transients are mitigated by the new system as effectively and reliability [reliably] as the existing system, the consequences of these transients have not changed. Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

LGS Modification P00224 uses digital processing with software (firmware) control for the main signal processing part of the modification. The remainder of the equipment in the modification uses conventional equipment similar to the current system (e.g., penetrations, cables, interface panels).

The digital equipment has ''control'' processing points and software-controlled digital processing where as the current system has analog and discrete component processing. The result is that the specific failures of hardware and potential software common cause failures are different from the current system. The effects of software common cause failure are mitigated by hardware design and system architecture, but are of a ''different type'' of failure than those evaluated in the LGS Updated Final Safety Analysis Report (UFSAR). Therefore, the replacement system may have a malfunction of a different type from those evaluated in the LGS UFSAR[. . .] However, when these PRNM failures are evaluated at the system level, there are no new effects.

LGS Modification P00224 involves equipment that is intended to detect the symptoms of certain transients and accidents and initiate mitigating action. The worst case failure of the equipment involved in the modification is a failure to initiate mitigating action (scram), but no failure can cause an accident. This is unchanged from the current system. Software common cause failures could result in the system failing to perform its safety function, but this possibility is addressed in Section 1, above. In that case, it might fail to initiate action to mitigate the consequences of an accident, but would not cause one. No new system level failure modes are created with the PRNM System.

Therefore, LGS Modification P00224 does 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 the margin of safety.

The PRNM System response time and operator information is either maintained or improved over the current Power Range Monitor System.

The PRNM System has improved channel trip accuracy compared to the current system and meets or exceeds system requirements assumed in setpoint analysis. The channel response time exceeds the requirements. The channel indicated accuracy is improved over the current system and meets or exceeds all of the system requirements.

The PRNM System was developed to detect the presence of thermal-hydraulic instabilities and automatically initiate the necessary corrective actions to suppress the oscillations prior to violating the Minimum Critical Power Ratio (MCPR) Safety Limit. The NRC has reviewed and approved the PRNM Licensing Topical Report (LTR) concluding that the PRNM System will provide the intended protection.

Therefore, LGS Modification P00224 does not result in a significant reduction in the margin of safety.

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

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

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* September 9, 1996, as supplemented on June 6, 1997, and June 7, 1999.

*Description of amendment request:* This application for amendment to the Indian Point 3 Technical Specifications (TSs) proposes to revise TS Section 6 to delete requirements for Plant Operating Review Committee review of the fire protection program and implementing procedures.

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

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

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

The proposed changes delete the Plant Operating Review Committee (PORC) review of changes to the fire protection program and implementing procedures. The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the changes do not degrade the performance of any safety system assumed to function in the accident analysis. Consequently, there is no effect on the probability or consequences of an accident.

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

No physical changes to the plant or changes to equipment operating procedures are proposed. The changes are administrative and will not have any direct effect on equipment important to safety. Therefore the changes cannot create the possibility of a new or different kind of accident.

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

Adequacy of the fire protection program and implementing procedures is assured by the fire protection license condition, the procedure review and approval process implemented by Amendment 159, the provisions of 10 CFR 50.59, and inspections and audits performed under the cognizance of the SRC [Safety Review Committee]. Consequently, deleting PORC's responsibility for review of the fire protection program and implementing procedure will not degrade the fire protection program. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

*Attorney for licensee:* Mr. David E. Blabey, 10 Columbus Circle, New York, New York 10019.

*NRC Section Chief:* Sheri R. Peterson.

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

*Date of amendment request:* November 8, 1999 (PCN 454).

*Description of amendment requests:* The licensee proposed to revise Surveillance Requirement (SR) 3.8.1.18 of Technical Specification (TS) 3.8.1, ''A.C. Sources-Operating.'' Currently, SR 3.8.1.18 reads: Verify interval between each sequenced load block is within plus or minus 10% of design interval for each emergency and shutdown load programmed time interval load sequence. The licensee proposed to revise the SR to read: Verify the timing of each sequenced load block is within its timer setting plus or minus 10% or plus or minus 2.5 seconds, whichever is greater, with the exception of the 5 second load group which is minus 0.5, plus 2.5 seconds, for each programmed time interval load sequence.

*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 any accident previously evaluated?

Response: No.

The proposed change would expand the current surveillance acceptance criteria to more accurately reflect the characteristics of the installed plant equipment. The diesel generators (DG's) have sufficient capacity to maintain adequate voltage and frequency during load sequencing with the expanded tolerance. The overall Engineered Safety Features (ESF) response times in the Technical Specifications and safety analyses are maintained even though the timer tolerance is increased. Therefore, the consequences of any accident previously evaluated are not increased. The DG load sequence timers are not of themselves a credible initiator of any accident, so the probability of an accident has not been increased. The timers will function acceptably to support the equipment needed for accident mitigation, so the consequences of an accident are not increased. Therefore, the probability or consequences of any accident previously evaluated are not increased.

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?

Response: No.

This amendment request does not involve any change to plant equipment or operation. In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DG's in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident such as a loss of coolant accident. Increasing the timer tolerance will 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?

Response: No.

This amendment does not change the manner in which safety limits, limiting safety settings, or limiting conditions for operations are determined. The actual response times have not been altered by this amendment. Therefore, operation of equipment will not be affected. Accordingly, this amendment will not involve a significant reduction in a margin of safety.

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

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

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* November 12, 1999 (PCN 505).

*Description of amendment requests:* The licensee proposed to revise Technical Specification (TS) 5.5.2.13, ''Diesel Fuel Oil Testing Program.'' Specifically, the following changes are proposed:

1. The at least once per 92 days test is deleted for water and sediment,

American Petroleum Institute (API) gravity or an absolute specific gravity, and kinematic viscosity for the diesel fuel oil in the Emergency Diesel Generator fuel oil storage tanks. The requirement to test these properties prior to addition of new fuel to the storage tank remains unchanged.

2. A requirement is added to test new fuel oil prior to addition to the storage tank to verify that the flash point is within limits.

3. A requirement is added to test new fuel oil within 31 days of delivery for ''other properties for ASTM [American Society for Testing and Materials] 2D fuel.''

4. The acceptance criteria for the properties listed, with the exception of the particulate criterion, are replaced with the phrase ''within limits.'' The statement which requires sampling in accordance with ASTM–D4057–81 is deleted. Acceptance criteria and reference to the applicable standard for sampling are currently provided in the Bases for Surveillance Requirement 3.8.3.3.

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

Response: No.

This change is an administrative change to make Technical Specification (TS) 5.5.2.13, ''Diesel Fuel Oil Testing Program,'' consistent with the existing Bases for Surveillance Requirement (SR) 3.8.3.3. The specific changes are:

1. The at least once per 92 days diesel fuel oil test is deleted for water and sediment, American Petroleum Institute (API) gravity or an absolute specific gravity, and kinematic viscosity. The requirement to test these properties prior to addition of new fuel to the storage tank remains unchanged.

2. A requirement is added to test new fuel oil prior to addition to the storage tank to verify that the flash point is within limits.

3. A requirement is added to test new fuel oil within 31 days of delivery for ''other properties for ASTM 2D fuel.''

4. The acceptance criteria for the properties listed, with the exception of the particulate content, are replaced with the phrase ''within limits.'' Acceptance criteria are currently provided in the Bases for Surveillance Requirement 3.8.3.3.

These changes are all consistent with the existing Bases for SR 3.8.3.3 and NUREG 1432.

Therefore, this change does not involve a significant increase in the probability or consequences of an 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?

Response: No.

This change is an administrative change to make TS 5.5.2.13, ''Diesel Fuel Oil Testing Program,'' consistent with the existing Bases for Surveillance Requirement 3.8.3.3.

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

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

Response: No.

This change is an administrative change to make TS 5.5.2.13, ''Diesel Fuel Oil Testing Program,'' consistent with the existing Bases for Surveillance Requirement 3.8.3.3.

Therefore, there will be no significant reduction in a margin of safety as a result of this change.

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

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

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* April 19, 1999, as supplemented by letter dated November 1, 1999.

*Description of amendment request:* The proposed change would revise Surveillance Requirement (SR) 3.3.5.2 and associated Bases to allow the loss of voltage and degraded voltage trip setpoints to be treated as nominal values in the same manner as the trip setpoints for the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation. The November 1, 1999, letter removes a note proposed in the April 19, 1999, amendment request. This revision does not change the scope of the April 19, 1999, application and the initial proposed no significant hazards consideration.

*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?

No. The proposed change affects only the presentation of the trip setpoints for loss of voltage and degraded voltage in SR 3.3.5.2 in the VEGP Units 1 and 2 TS [Technical Specifications]. The calibration of the channels whose setpoints are specified in SR 3.3.5.2 will continue to be performed in a manner consistent with the setpoint methodology used to determine the trip setpoints. There will be no adverse effect on the ability of those channels to perform their safety functions as assumed in the safety analyses. Since there will be no adverse effect on the trip setpoints or the instrumentation associated with those trip setpoints, there will be no increase in the probability of any accident previously evaluated. Similarly, since the ability of the instrumentation to perform its safety function is not adversely affected, there will be no increase in the consequences of any accident previously evaluated.

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

No. The proposed change affects only the presentation of the trip setpoint requirements of SR 3.3.5.2. Plant operation will not be changed, and the response of safety related equipment as assumed in the accident analyses would not be adversely affected. Therefore, the proposed change does not involve a new or different kind of accident than any previously evaluated.

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

No. As described above, the loss of voltage and degraded voltage instrumentation will remain capable of performing its safety function as assumed in the accident analyses. The treatment of trip setpoints as nominal values is consistent with the methodology used to establish those setpoints. As such, margin is not affected by the proposed change.

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:* September 8, 1999, as supplemented by letter dated November 9, 1999. The September 8, 1999, application was originally noticed in the **Federal Register** on November 3, 1999 (64 FR 59806).

*Description of amendment request:* The proposed amendments would revise Technical Specification 3/4.8.1, ''A.C. Sources, Operating,'' and associated Bases, by relocating the 18-month surveillance to subject the standby diesel generator to inspections, in accordance with procedures prepared in conjunction with its manufacturer's recommendations, to the Technical Requirements Manual.

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

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

The proposed change moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the Technical Requirements Manual (TRM). The change does not result in any hardware or operating procedure changes. The requirement being removed from the Technical Specifications is not the initiator of any analyzed event. The TRM is maintained using the provisions of 10 CFR 50.59. Since any changes will be evaluated per 10 CFR 50.59, no significant increase in the probability or consequences of an accident previously evaluated will be allowed without prior NRC approval. Therefore, the 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 change moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the TRM. The change does not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The change does not impose different requirements. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change moves the requirement to perform manufacturer's recommended inspections of the Standby Diesel Generators from the Technical Specifications to the TRM. The change does not reduce the margin of safety since the location of details has no impact on any safety analysis assumptions. In addition, the requirement being transposed from the Technical Specification to the TRM is the same as the existing Technical Specification. Also, the TRM is maintained using the provisions of 10 CFR 50.59. Since any changes will be evaluated per 10 CFR 50.59, no significant reduction in a margin of safety will be allowed without prior NRC approval.

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.

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

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

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

*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:* November 5, 1999.

*Description of amendment requests:* The proposed license amendments would revise Technical Specification (T/S) Surveillance Requirement 4.5.1.c to require verification that power is removed from each emergency core cooling system accumulator isolation valve operator instead of verification that each accumulator isolation valve breaker is removed from the circuit. In addition, the proposed license amendments would revise T/S 3.5.1 to change ''pressurizer pressure'' to ''reactor coolant system pressure'' in the applicability and action statement requirements. The Bases for T/S 3/4.5.1 will also be revised to reflect both changes. Additionally, administrative changes are proposed to the page format.

*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 of occurrence or consequences of an accident previously evaluated?

The ECCS [emergency core cooling system] accumulators are used to mitigate the consequences of an accident after the event has occurred and do not initiate any accident previously evaluated. Demonstrating how power is removed from the valve operator does not initiate an accident. Inadvertently closing the valves cannot initiate an accident. Therefore, there is no significant increase in the probability of occurrence of an accident previously evaluated.

The ECCS accumulators will still perform their function of injecting borated water into the reactor coolant loops following a large break loss-of-coolant accident, as described in Section 14.3.1 of the Updated Final Safety Analysis Report (UFSAR). A spurious closure of an accumulator outlet isolation valve is not a credible event. Performing T/S Surveillance Requirement 4.5.1.c provides assurance that one of the two actions required for spurious closure of the valve is precluded. The proposed change to the surveillance continues to provide assurance that power will be removed from each accumulator isolation valve operator so that the valves remain open. The consequences of accidents previously evaluated remained bounded because the accumulators will still function as assumed in the UFSAR accident analysis. Therefore, there is no significant increase in the consequences of any accident previously evaluated.

Changing ''pressurizer pressure'' to ''RCS [reactor coolant system] pressure'' has no significant effect on the applicability of the T/S requirements. RCS pressure and pressurizer pressure instrumentation measure a similar parameter in the primary coolant system. Since the RCS is a closed-loop fluid system, pressure instruments should indicate approximately the same value. There is no significant difference between the instrument readings because they are corrected for range, height, and accuracy. There is no significant change in the margin of pressure between when the accumulators are required to be aligned at 1000 psig and the upper limit specified in T/S 3.5.1.d of 658 psig.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, the proposed changes do not increase the probability of occurrence 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 to T/S 3/4.5.1 and the associated Bases do not involve any physical changes to the plant, but do change the way the plant is operated by changing the method for ensuring spurious closure of the accumulator isolation valve will not occur. The proposed change to T/S Surveillance Requirement 4.5.1.c does not create any new operator actions. The position of the accumulator isolation valve remains open in Modes 1, 2, and 3 with RCS pressure greater than 1000 psig, which meets its design safety function. The proposed change does not increase the possibility of the accumulator valve repositioning. In order for repositioning to happen, the operator must close the molded-case circuit breaker coupled with either an active single failure or deliberate operator action in the control room. The proposed change of verifying that power is removed from the accumulator isolation valve provides the same level of protection. Two positive actions are required for the accumulator isolation valve to reposition.

The proposed format changes are administrative and have no impact on plant operation.

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?

T/S Surveillance Requirement 4.5.1.c provides requirements that ensure that a single action will not cause an inadvertent closure of the accumulator isolation valves. The proposed change continues to ensure that two positive actions, an operator action to restore the breaker and a single failure, are required for valve closure.

Changing ''pressurizer pressure'' to ''RCS pressure'' does not impact operation of the accumulators. The proposed changes do not impact the nitrogen cover pressure as stated in T/S 3.5.1.c. The accumulators would not be expected to inject borated water until RCS pressure lowers to 658 psig (the upper limit specified in T/S 3.5.1.d). The change does not affect when this would occur after an accident. Therefore, changing ''pressurizer pressure'' to ''RCS pressure'' has no impact on plant operation.

The proposed format changes are administrative and have no impact on plant operation.

Therefore, there is no 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 requests involves no significant hazards consideration.

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

*NRC Section Chief:* Claudia M. Craig.

## Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

*Arizona Public Service Company, et al., Docket No. STN 50–528, Palo Verde Nuclear Generating Station, Unit No. 1, Maricopa County, Arizona*

*Date of application for amendment:* October 8, 1999, as supplemented October 29, 1999.

*Brief description of amendment:* The amendment revises Surveillance Requirement 3.8.4.8 of Technical Specification 3.8.4, to allow the licensee to forego the performance of this surveillance until entry into MODE 4 coming out of the ninth refueling outage for Unit 1.

*Date of issuance:* November 19, 1999.

*Effective date:* November 19, 1999.

*Amendment No.:* 121.

*Facility Operating License No. NPF–41:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: October 19, 1999 (64 FR 56369).

The October 29, 1999, supplement provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 19, 1999.

No significant hazards consideration comments received: No

*Commonwealth Edison Company, Docket No. 50–373, LaSalle County Station, Unit 1, LaSalle County, Illinois*

*Date of application for amendment:* July 7, 1999, as supplemented on October 14, 1999.

*Brief description of amendment:* The amendment revised Section 2.1 of the Technical Specifications to reflect a change in the Minimum Critical Power Ratio.

*Date of issuance:* November 9, 1999.

*Effective date:* Immediately, to be implemented prior to the startup of Cycle 9.

*Amendment No.:* 137.

*Facility Operating License No. NPF–11:* The amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: August 11, 1999.

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

No significant hazards consideration comments received: No.

*Duquesne Light Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania*

*Date of application for amendment:* January 29, 1998, as supplemented by letters dated November 9, 1998, and June 14, 1999.

*Brief description of amendment:* This amendment authorized changes to the Beaver Valley Power Station, Unit No. 2 (BVPS–2) Updated Final Safety Analysis Report (UFSAR). The amendment authorizes changes to the UFSAR to reflect revisions to the radiological dose calculations for the locked rotor accident analysis. This revision of the calculation was performed in order to incorporate more conservative assumptions than those used in the previous analysis for a postulated locked rotor event.

These changes are not the result of hardware changes to the plant or any change in operating practices. They reflect revised analysis results only and allow revision of the licensing basis to reflect conservative assumptions used in the revised analyses.

The June 14, 1999, letter withdrew a portion of the amendment which would have revised the UFSAR description of the small-break loss-of-coolant accident radiological consequences.

*Date of issuance:* November 18, 1999.

*Effective date:* As of the date of issuance.

*Amendment No:* 103.

*Facility Operating License No. NPF–73.* Amendment approved changes to the UFSAR.

*Date of initial notice in* **Federal Register**: March 11, 1998 (63 FR 11919).

The November 9, 1998, and June 14, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1999.

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:* July 29, 1999, as supplemented by letters dated August 6, 1999, October 14, 1999, and October 26, 1999.

*Brief description of amendment:* The proposed change to the Arkansas Nuclear One, Unit No. 2 Technical Specifications would allow the performance of a special inspection of the steam generator tubes during an upcoming mid-cycle outage. This mid-cycle outage is planned for the purpose of performing inspections in selected areas of the steam generator tube bundle where previous inspections have revealed tube degradation. The proposed change would limit the initial inspection scope to these identified areas and includes scope expansion criteria to address unexpected results.

*Date of issuance:* November 5, 1999.

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

*Amendment No.:* 210.

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

*Date of initial notice in* **Federal Register**: October 6, 1999 (64 FR 54375).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 5, 1999.

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:* May 6, 1999.

*Brief description of amendment:* The amendment incorporates the Technical Specification changes necessary for redefining the minimum critical power ratio safety limit for Cycle 11 operation with a mixed core of Siemens Power Corporation fuel and General Electric fuel.

*Date of issuance:* November 17, 1999.

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

*Amendment No:* 140.

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

*Date of initial notice in* **Federal Register**: August 25, 1999 (64 FR 46434).

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

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:* July 26, 1999.

*Brief description of amendment:* This amendment—

(1) Relocates the requirements in TS 3/4.3.3.2, ''Instrumentation—Incore Detectors,'' TS 3/4.3.3.9, ''Instrumentation—Waste Gas System Oxygen Monitor,'' and TS 3/4.4.4.7, ''Reactor Coolant System—Chemistry,'' to the Davis-Besse Nuclear Power Station (DBNPS) Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM);

(2) Revises TS 3/4.11.2, ''Radioactive Effluents—Explosive Gas Mixture,'' to reflect the relocation of TS 3/4.3.3.9;

(3) Revises the requirements of TS 3/4.4.6.1, ''Reactor Coolant System Leakage—Leakage Detection Systems,'' to require one monitor (gaseous or particulate) of the containment atmosphere radioactivity monitoring systems to be operable, rather than requiring both systems to be operable simultaneously; and

(4) Revises TS 3/4.3.3.1, ''Radiation Monitoring Instrumentation,'' to be consistent with the revision to TS 3/4.4.6.1.

*Date of issuance:* November 16, 1999

*Effective date:* November 16, 1999.

*Amendment No.:* 234.

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

*Date of initial notice in* **Federal Register**: August 25, 1999 (64 FR 46436).

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

No significant hazards consideration comments received: No

*NASA Aeronautics Space Administration (NASA), Docket No. 50–30, NASA Test Reactor, Erie County, Ohio*

*Date of application for amendment:* March 25, 1999, as supplemented on August 10, 1999.

*Brief description of amendment:* This amendment changes Lewis Research Center (LeRC) to Glenn Research Center (GRC).

*Date of issuance:* November 16, 1999.

*Effective Date:* November 16, 1999.

*Amendment No:* 10.

*Facility License No. TR–3:* The amendment changes facility name.

*Date of initial notice in* **Federal Register**: October 6, 1999 (64 FR 54377).

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

No significant hazards consideration comments received: No.

*Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York*

*Date of application for amendment:* November 16, 1998, as supplemented June 21, 1999.

*Brief description of amendment:* Amendment changes Technical Specifications to limit reactor power oscillations during a reactor trip and allows operation in the Extended Load Line Limit Analysis region of the power/flow operating curve.

*Date of issuance:* September 21, 1999.

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

*Amendment No.:* 168.

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

*Date of initial notice in* **Federal Register**: December 30, 1998 (63 FR 71968) as corrected January 27, 1999 (64 FR 4148).

The June 21, 1999, letter provided supporting information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 21, 1999.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* September 29, 1998, as supplemented by letters dated March 8 and April 7, 1999.

*Description of amendment request:* To revise Facility Operating License No. NPF–86 to reflect the transfer of the license, to the extent held by Montaup Electric Company, to Little Bay Power Corporation.

*Date of issuance:* November 19, 1999.

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

*Amendment No.:* 65.

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

*Date of initial notice in* **Federal Register**: December 14, 1998 (63 FR 68801). The March 8 and April 7, 1999 supplements provided clarifying information and did not change the staff's proposed no significant hazards determination. The Commission received comments which were addressed in the staff's Safety Evaluation dated August 3, 1999. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 1999.

No significant hazards consideration comments received: Yes.

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

*Date of application for amendments:* April 19, 1999, as supplemented August 25, October 14, and November 3, 1999.

*Brief description of amendments:* The amendment deletes most of the current Technical Specifications to implement the Permanently Defueled Technical Specification. Portions of the April 19, 1999, request related to fuel storage pool water level, crane operability, and crane travel with a spent fuel cask will be addressed at a later date.

*Date of issuance:* November 9, 1999.

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

**Amendment No.: 106.**

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

*Date of initial notice in* **Federal Register**: June 30, 1999 (64 FR 35208).

The August 25, 1999, letter provided clarifying information that did not change the scope of the April 19, 1999, application and the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No

*Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone*

*Nuclear Power Station, Unit No. 3, New London County, Connecticut*

*Date of application for amendment:* May 7, 1998, as supplemented January 22, 1999.

*Brief description of amendment:* The amendment revises the licensing basis to address the addition of the dose from the Refueling Water Storage Tank back leakage into the design basis loss-of-coolant accident analysis and Chapter 15 of the Final Safety Analysis Report.

*Date of issuance:* November 4, 1999.

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

*Amendment No.:* 176.

*Facility Operating License No. NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: July 1, 1998 (63 FR 35991). The January 22, 1999, supplement provided clarifying information that did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 4, 1999.

No significant hazards consideration comments received: No

*Northeast Nuclear Energy Company, et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut*

*Date of application for amendment:* August 5, 1999.

*Brief description of amendment:* The amendment corrects editorial errors in the Technical Specifications Sections 3.8.3.2, 4.6.2.1, 4.8.1.1, and 4.9.12. The amendment also corrects minor editorial and reference errors in Bases Sections B 3/4.3.2, B 3/4.4.11, B 3/4.6.1.2, and B 3/4.8.4.

*Date of issuance:* November 15, 1999.

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

*Amendment No.:* 177.

*Facility Operating License No. NPF–49:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: September 8, 1999 (64 FR 48858).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 1999.

No significant hazards consideration comments received: No.

*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 application for amendments:* December 29, 1998, as supplemented by letters dated July 30 and October 12, 1999.

*Brief description of amendments:* The amendments revise Technical Specifications (TS) 6.9.1.8, ''Core Operating Limits Report,'' of the current TSs and TS 5.6 of the improved TSs, to allow the use of NRC approved addenda to WCAP–10054–P–A, ''Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code,'' August 1985, to determine core operating limits. The improved TSs were issued in Amendment Nos. 135 for Diablo Canyon Power Plant, Units 1 and 2 dated May 28, 1999, but have not yet been implemented.

*Date of issuance:* November 15, 1999.

*Effective date:* November 15, 1999, and shall be implemented within 90 days from the date of issuance.

*Amendment Nos.:* Unit 1—136; Unit 2–136.

*Facility Operating License Nos. DPR–80 and DPR–82:* The amendments revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: April 21, 1999 (64 FR 19562). The July 30 and October 12, 1999, supplemental letters provided additional clarifying information and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 15, 1999.

No significant hazards consideration comments received: No.

*PECO Energy Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania.*

*Date of amendment request:* January 12, 1999, as supplemented January 29, March 10, and September 20, 1999.

*Description of amendment request:* This amendment revised Technical Specifications (TSs) Section 3/4.4.2, ''Safety/Relief Valves,'' and TS Bases Sections B 3/4.4.2, B 3/4.5.1 and B 3/4.5.2 to increase the allowable as-found main steam safety relief valve (SRV) code safety function lift setpoint tolerance from plus or minus 1% to plus or minus 3%. Also, the required number of operable SRVs in operational conditions 1, 2, and 3 will be increased from 11 to 12.

*Date of issuance:* November 10, 1999.

*Effective Date:* As of date of issuance and shall be implemented prior to completion of the spring 2000 refueling outage for Limerick Generating Station, Unit 1.

*Amendment No:* 137.

*Facility Operating License No. NPF–39.* The amendment revises the Technical Specifications.

*Date of initial notice in* **Federal Register**: February 24, 1999 (64 FR 9194).

The January 29, March 10, and September 20, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 10, 1999.

No significant hazards consideration comments received: No.

*PECO Energy Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania.*

*Date of application for amendment:* June 7, 1999.

*Brief description of amendment:* The amendment revised the technical specifications (TSs) to reflect the permanent deactivation in the closed position of the ''wet'' instrument reference leg isolation valve HV–61–102. Specifically, TS Table 3.6.3.1, ''Primary Containment Isolation Valve,'' and its associated notations were revised to reflect this current plant configuration.

*Date of issuance:* November 18, 1999.

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

*Amendment No.:* 138.

*Facility Operating License No. NPF–39.* This amendment revised the TSs.

*Date of initial notice in* **Federal Register**: October 6, 1999 (64 FR 54380).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 18, 1999.

No significant hazards consideration comments received: No.

*PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50–277 and 278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania*

*Date of application for amendments:* December 24, 1998, as supplemented May 25 and September 27, 1999.

*Brief description of amendments:* These amendments revise Technical Specification (TS) Table 3.3.8.1–1 related to loss of power instrumentation set points and limits of allowable values for the 4 kV emergency buses.

*Date of issuance:* November 16, 1999.

*Effective date:* These license amendments are effective as of their date of issuance. Phase 1 applies to Functions 2 and 3 in TS Table 3.3.8.1–1 and shall be implemented within 30 days of the date of issuance of the amendment. Phase 2 applies to Functions 4 and 5 in TS Table 3.3.8.1–1 and shall be implemented no later than March 1, 2000. Note (a) shall be implemented within 30 days of the date of issuance of the amendment and shall be voided upon completion of modification 96–01511, but no later than March 1, 2000.

*Amendments Nos.:* 230 and 235.

*Facility Operating License Nos. DPR–44 and DPR–56:* The amendments revised the Technical Specifications. The May 25 and September 27, 1999, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

*Date of initial notice in* **Federal Register**: May 5, 1999 (64 FR 24199).

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* October 14, 1997, as supplemented July 23, 1998, December 3, 1998, February 25, 1999, and September 29, 1999.

*Brief description of amendment:* The amendment revises Technical Specifications to permit use of additional spent fuel storage racks.

*Date of issuance:* November 10, 1999.

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

*Amendment No.:* 256.

*Facility Operating License No. DPR–59:* Amendment revised the Technical Specifications.

*Date of initial notice in* **Federal Register**: August 24, 1998 (63 FR 45096).

The July 23, 1998, December 3, 1998, February 25, 1999, and September 29, 1999, applications provided supplemental information that did not affect the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 10, 1999.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* November 14, 1997, as supplemented on August 25, 1999.

*Brief description of amendments:* The amendments revise the TSs to make administrative and editorial changes to correct errors in the TSs that have either existed since initial issuance or were introduced during subsequent changes. In addition, surveillance requirements are added that should have been incorporated within the TSs when the applicable amendment to the TSs was approved by the NRC.

*Date of issuance:* November 2, 1999.

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

*Amendment Nos.:* 225 and 206.

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

*Date of initial notice in* **Federal Register**: December 17, 1997 (63 FR 66141). The August 25, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 1999.

No significant hazards consideration comments received: No.

*Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California*

*Date of application for amendments:* March 18, 1996, as supplemented April 28, 1997, and February 16, 1999.

*Brief description of amendment:* The amendment authorizes changes to the design-basis accident analysis (postulated cask drop accident) to be incorporated into the Defueled Safety Analysis Report (DSAR) and revises the Permanently Defueled Technical Specifications to reflect the changes to the cask drop analysis.

*Date of issuance:* November 12, 1999.

*Effective date:* November 12, 1999, with the Technical Specifications to be implemented within 30 days. Implementation also includes incorporation of the changes into the DSAR at the next update of the DSAR in accordance with the schedule in 10 CFR 50.71(e).

*Amendment No.:* 127.

*Facility Operating License No. DPR–54:* The amendment revised the Technical Specifications and the Defueled Safety Analysis Report.

*Date of initial notice in* **Federal Register**: August 25, 1999 (64 FR 46442).

The April 28, 1997, and February 16, 1999, supplements provided additional clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 12, 1999.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* October 20, 1998 (PCN 485), as supplemented August 13, 1999.

*Brief description of amendments:* The amendments revise Technical Specification 3.3.9 by adding a surveillance requirement for response time testing for the control room isolation signal.

*Date of issuance:* November 15, 1999.

*Effective date:* November 15, 1999, to be implemented within 30 days of issuance.

*Amendment Nos.:* Unit 2—160; Unit 3—151.

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

*Date of initial notice in* **Federal Register**: October 12, 1999 (64 FR 55311).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 15, 1999.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* May 7, 1998, as supplemented by letters dated May 20, June 16, September 30, October 20, and October 21, 1999.

*Brief description of amendments:* The amendments changed the Technical Specifications (TSs) to reflect reactor coolant system flow differences between the existing Model E and replacement Model ®94 steam generators (SGs) by adding a new flow rate requirement to TS 3.2.5, Departure from Nucleate Boiling (DNB) Parameters, that is applicable to the Model ®94 SGs. Related changes to Bases 3/4.2.5, DNB Parameters, were also made. The licensee withdrew all changes proposed in the May 7, 1998, application that were superseded by the previously approved amendments 115/103 dated September 2, 1999.

*Date of issuance:* November 8, 1999.

*Effective date:* November 8, 1999.

*Amendment Nos.:* Unit 1—117; Unit 2—105.

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

*Date of initial notice in* **Federal Register**: July 1, 1998 (63 FR 35996).

The May 20, June 16, September 30, October 20, and October 21, 1999, supplements provided additional clarifying information. The September 30, 1999, supplement also provided updated TS pages. This information was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of amendment request:* August 31, 1998, as supplemented by letters dated April 19, August 18, and October 21, 1999.

*Brief description of amendments:* The amendments revised Technical Specification 3/4.4.9.3 by revising the cold overpressure mitigation curve to accommodate the replacement steam generators and by adding two surveillances (for the centrifugal charging pumps and the emergency core cooling system accumulators) to ensure the operability of the cold overpressure mitigation system.

*Date of issuance:* November 9, 1999.

*Effective date:* November 9, 1999, to be implemented within 30 days.

*Amendment Nos.:* Unit 1—118; Unit 2—106.

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

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

The October 21, 1999, supplement provided a revised implementation date. This information was within the scope of the original application and **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

**Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)**

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

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By January 3, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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: Rulemakings and Adjudications Staff or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

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

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

*Date of amendment request:* October 29, 1999, as supplemented November 2, 1999.

*Description of amendment request:* The amendment revises the Technical Specification administrative controls regarding the containment leak rate testing program and the core operating limits report. These changes are necessary to reflect changes in the accident analyses and core design methodologies for the next operating cycle.

*Date of issuance:* November 15, 1999.

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

*Amendment No.:* 188.

*Facility Operating License No. DPR–20:* Amendment revises the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes. The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration, and requested that any comments on the proposed no significant hazards consideration be provided to the staff by close of business November 12, 1999. The notice was published in the Herald Palladium on November 6–8, 1999. No public comments were received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated November 15, 1999.

*Attorney for licensee:* Judd L. Bacon, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

*NRC Section Chief:* Claudia M. Craig.

Dated at Rockville, Maryland, this 23rd day of November 1999.

For the Nuclear Regulatory Commission.

**Suzanne C. Black,**

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

[FR Doc. 99–31037 Filed 11–30–99; 8:45 am]

**BILLING CODE 7590–01–P**

## POSTAL RATE COMMISSION

### Tour of Printing and Processing Plants

**AGENCY:** Postal Rate Commission.

**ACTION:** Notice of Commission visit.

**DATES:** The visits are scheduled for December 6–8, 1999.

**FOR FURTHER INFORMATION CONTACT:** Stephen L. Sharfman, General Counsel, Postal Rate Commission, Suite 300, 1333 H Street, NW., Washington, DC 20268–0001, 202–789–6820.

**SUPPLEMENTARY INFORMATION:** Members of the Postal Rate Commission will visit the R.R. Donnelley printing plant at Spartanburg, South Carolina on the afternoon of Monday, December 6, 1999. The Commission will discuss logistics and support issues, and problems with and procedures for preparation of mail for dropshipping. On the morning of Tuesday, December 7, 1999, the group will tour the BMG fulfillment facility in Duncan, South Carolina, and discuss mailing practices that incorporate the use of multiple subclasses and services by a major music club. That evening, the group will observe operations at the Orlando, Florida terminal facility used by members of the Florida Gift Fruit Shippers Association (FGFSA) to prepare items for shipment to distant postal facilities.

On Wednesday, December 8, 1999 the group will tour the packinghouse operation of a shipper-member of FGFSA to get a complete understanding of parcel movement from producers to consumers using the Postal Service delivery network, and then meet with several shippers to obtain a balanced picture of the varying needs of different sized operations. Finally, during the evening of December 8, the group will observe the operation of the Orlando Priority Mail processing center operated for the Postal Service by Emery.

Dated: November 24, 1999.

**Margaret P. Crenshaw,**

*Secretary.*

[FR Doc. 99–31170 Filed 11–30–99; 8:45 am]

**BILLING CODE 7710–01–P**

## SECURITIES AND EXCHANGE COMMISSION

**[Rel. No. IC–24176; 812–11402]**

### INVESCO Bond Funds, Inc., et al.; Notice of Application

November 24, 1999.

**AGENCY:** Securities and Exchange commission (''SEC'').

**ACTION:** Notice of application under section 6(c) of the Investment Company Act of 1940 (''Act'') for an exemption from sections 18(f) and 21(b) of the Act, under section 12(d)(1)(J) of the Act for an exemption from section 12(d)(1) of the Act, under sections 6(c) and 17(b) of the Act for an exemption from sections 17(a)(1) and 17(a)(3) of the Act, and under section 17(d) of the Act and rule 17d–1 under the Act to permit certain joint arrangements.

**SUMMARY OF APPLICATION:** Applicants request an order that would permit certain registered investment management companies to participate in a joint lending and borrowing facility.

**APPLICANTS:** INVESCO Bonds Funds, Inc., INVESCO Combination Stock and Bond Funds, Inc., INVESCO Global Health Sciences Fund, INVESCO International Funds, Inc., INVESCO Money Market Funds, Inc., INVESCO Sector Funds, Inc., INVESCO Speciality Funds, Inc., INVESCO Stock Funds, Inc., INVESCO Treasurer's Series Funds, Inc., and INVESCO Variable Investment Funds, Inc. (collectively, the ''Companies''), INVESCO Funds Group, Inc. (''INVESCO Funds Group,'' and together with any entity controlling, controlled by, or under common control with INVESCO Funds Group, ''INVESCO''), and any other registered open-end investment company advised by INVESCO (together with the Companies, the ''Funds'').

**FILING DATES:** The application was filed on November 13, 1998, and amended on October 15, 1999. Applicants have agreed to file an additional amendment during the notice period, the substance of which is reflected in this notice.

**HEARING OR NOTIFICATION OF HEARING:** An order granting the application will be