NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures; Revised

The ACRS Subcommittee on Planning and Procedures scheduled to start at 1 p.m. on *Tuesday, July 11, 2000, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland has been changed to start at 9 a.m.* Notice of this meeting was published in the **Federal Register** on Monday, June 26, 2000 (65 FR 39446). All other items pertaining to this meeting remain the same as previously published.

FOR FURTHER INFORMATION CONTACT: Dr. John T. Larkins, cognizant ACRS staff person (telephone: 301/415–7360) between 7:30 a.m. and 4:15 p.m. (EDT).

Dated: July 6, 2000.

Howard J. Larson,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–17623 Filed 7–11–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATES: Weeks of July 10, 17, 24, 31, August 7, and 14, 2000.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 10

Monday, July 10

1:25 p.m.—Affirmation Session (Public Meeting)
- a: Rulemaking to Modify the Event Reporting Requirements for Power Reactor in 10 CFR 50.72 and 50.73 and for Independent Spent Fuel Storage Installations (ISFSI) in 10 CFR 72.216
- b: Final Rule: 10 CFR Parts 30, 31, and 32—''Requirements for Certain Generally Licensed Industrial Devices Containing Byproduct Material'' and Related Change to the Enforcement Policy
- c: Hydro Resources, Inc. Petition for Review of LBP–99–18, LBP–99–19, and LBP–99–30

1:30 p.m.—Briefing on Proposed Export of High Enriched Uranium to Canada (Public Meeting)

Tuesday, July 11

9:30 a.m.—Discussion of Intragovernmental Issues (Closed–Ex. 4 and 9)

Week of July 17—Tentative

There are not meetings scheduled for the Week of July 17.

Week of July 24—Tentative

Tuesday, July 25

3:25 p.m.—Affirmation Session (Public Meeting), (If necessary)

Week of July 31—Tentative

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

Week of August 7—Tentative

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

Week of August 14—Tentative

Tuesday, August 15

9:25 a.m.—Affirmation Session (Public Meeting), (If necessary)

9:30 a.m.—Briefing on NRC International Activities (Public Meeting), (Contact: Ron Hauber, 301–415–2344)

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meeting 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, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: July 9, 2000.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 00–17780 Filed 7–12–00; 2:18 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 17, 2000, through June 30, 2000. The last biweekly notice was published on June 28, 2000 (65 FR 39956).

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

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

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

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period.

However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

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

By August 11, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: June 21, 2000. This request supplements an earlier application dated October 29, 1999, submitted by GPU Nuclear, Inc., which has since been adopted by AmerGen Energy Company, LLC.

Description of amendment request: The proposed amendment revises the Technical Specifications (TSs) to include: (1) The addition of operating limits for make-up tank (MUT) level and pressure; (2) the addition of surveillance requirements for the MUT pressure instrument channel; and (3) revision of the calibration frequency for the MUT level instrument channel from ''Not to exceed 24 months'' to ''Refueling interval (once per 24 months)'' along with other instruments (high pressure and low pressure injection (LPI) flow instruments and the borated water storage tank (BWST) level instrument) in the same table as appropriate. Associated Bases changes are also proposed. Minor editorial changes (such as updates to the Table of Contents and others) are also proposed. This revision to the original submittal reflects changes to proposed TS Figure 3.3–1 and adds an additional instrument to those for which a surveillance calibration frequency extension is requested.

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

A. *The proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.*

The changes included in this LCA [license change application] impose new requirements for MU/HPI [make-up/high pressure injection] system operation and testing and extension of calibration frequencies for the MUT level, HPI flow and LPI flow instruments and BWST level instrument. These changes could not result in initiation of any accident previously evaluated. Therefore, the probability of an accident could not be affected by changes to the MU/HPI and Decay Heat Removal (DHR) systems.

As described in the list of benefits for operation with MU/HPI cross-connect valves open, listed in section III.B above [section III.B, pages 5–6 of 14, of the June 21, 2000 supplement], the purpose of changing the operation of the MU/HPI system was to preclude the possibility of HPI pump damage. The addition of surveillance requirements for the MUT pressure instrument and the addition of LCO [limiting condition for operation] limits on MUT level and pressure along with appropriate action statements and required action times will ensure that gas entrainment of the MUT does not occur. The proposed change in instrument calibration frequencies will continue to maintain the required accuracy of the MUT level, HPI flow, LPI flow, and BWST level instruments.

Minor editorial changes are included in this request to improve clarity and readability of the T.S. [technical specifications] and could not adversely affect plant operation.

Therefore, the proposed changes will not adversely impact the reliability of the MU/HPI system and could not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

This LCA does not involve the addition of any new hardware. Along with minor editorial changes, the requested changes involve MU/HPI system operation and changes in instrument calibration frequency which have been reviewed in accordance with NRC guidance. Changes to MU/HPI System operation can only affect RCS [reactor coolant system] coolant inventory changes during operation and the ability to provide protection in the event of a Loss of Coolant Accident (LOCA). The full spectrum of LOCAs has been evaluated in the FSAR [Final Safety Analysis Report]. Therefore, no new accident scenarios have been created.

The additional controls on MUT level and pressure provided by this LCA will ensure that a malfunction of a different type, gas entrainment of the MU/HPI pumps, will not occur. These limits on MUT level and pressure ensure that the initial conditions assumed for ECCS operation are maintained. The TS limits maintain the accident analysis initial conditions such that no operator action is required to avoid gas entrainment during ECCS [emergency core cooling [system] operation with the postulated single failure as required by the TMI–1 licensing basis (Reference 14) [GPU Nuclear Safety Evaluation No. SE–000211–015, Revision 0, ''Operation with MU X–Connect Valves OPEN''].

Extension of the calibration frequencies for the HPI level, HPI flow, LPI flow, and BWST level will continue to maintain the accuracy of these instruments and could not create the potential for any new accident that has not been evaluated.

Minor editorial changes are included in this request to improve the clarity and readability of the TS and could not adversely affect plant operation.

Therefore, these [proposed] changes do not create the potential for any accident different from those that have been evaluated.

C. *These proposed changes do not involve a significant reduction in a margin of safety.*

This LCA includes changes to MU/HPI system operation and testing and an extension of the calibration frequency for certain instruments. The requested changes will serve to maintain the proper system initial conditions to ensure the ability of the MU/HPI system to provide protection in the event of a Loss of Coolant Accident (LOCA) and maintain the required instrument accuracy for the instruments where changes to a refueling interval frequency are being requested. NRC guidance for addressing the effect on increased surveillance intervals on instrument drift and safety analysis assumptions presented in GL [generic letter] 91–04 have been addressed in enclosure 1A [of the licensee's June 21, 2000 letter].

Minor editorial changes are included in this request to improve clarity and readability of the TS and could not adversely affect plant operation.

These changes, which are consistent with the TMI–1 licensing and design basis requirements, do not result in a degradation of safety related equipment, and therefore, do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis (paragraph 'B') against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below.

The licensee concluded that ''these [proposed] changes do not create the potential for any accident different from those that have been evaluated.'' This conclusion is worded slightly differently than the standard in 10 CFR 50.92 (''The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated'') that the licensee is required to analyze against pursuant to 10 CFR 50.91. Nevertheless, the licensee did state in its application that ''additional controls on MUT level and pressure provided by the proposed changes in this LCA will ensure that a malfunction of a different type, gas entrainment of the MU/HPI pumps, will not occur.'' These additional controls include a prohibited operating region which would require plant shutdown if not corrected.

The licensee further stated, that the portion of the TS Figure 3.3–1 related to NPSH [net positive suction head] has been deleted because operation of MU/HPI pump below the manufacturer's NPSH limits for a short period of time may affect pump performance while the NPSH shortfall exists, but would not render the pump inoperable. The licensee further stated that existing plant procedures will provide NPSH MUT pressure verses level operating limits that will ensure the recommended NPSH would be available for the NPSH limiting event, an HPI line break small-break LOCA. Based on the above, the staff has determined that the proposed changes and additional controls on MUT level and pressure would not create the possibility of a new or different kind of accident from any previously evaluated.

The licensee has determined that the proposed extension of the calibration frequencies for the HPI level, HPI flow, LPI flow, and BWST level, meets applicable staff guidance related to these proposed changes and will continue to maintain the accuracy of these instruments and could not, therefore, create the potential for any new accident that has not been evaluated. The staff has determined that the proposed extension of calibration frequencies would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed editorial changes are minor in nature, and are intended to improve the clarity and readability of the TSs, and would not create the possibility of a new or different kind of accident from any previously evaluated.

Based on this review, and the licensee's basis for its determination with respect to items ''A'' and ''C'' above, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Marsha Gamberoni.

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

Date of amendments request: June 6, 2000.

Description of amendments request: The proposed amendments would revise information in Figure 3.5.5–1, ''Minimum Required RWT Volume,'' in Technical Specification (TS) 3.5.5, ''Refueling Water Tank (RWT),'' of the TSs for the three units. The amendments are administrative changes to the figure that would (1) Relocate design bases information to the Bases of the TSs, (2) truncate the lower end of the RWT limit curve at 210 °F, (3) re-title the right-hand ordinate from ''minimum useful volume required in the RWT'' to ''RWT Volume,'' and (4) delete the two footnotes and the references to the footnotes.

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 in its application, which is presented below:

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

This proposed administrative change does not involve any changes to the design, operation, or maintenance of any structures[,] systems or components. The requirements in TS 3.5.5 for RWT operability will not be changed. This proposed amendment [for each unit] does not alter, degrade, or prevent actions described or assumed in an accident described in the PVNGS UFSAR [Updated Final Safety Analysis Report] from being performed. It will not alter any assumptions previously made in evaluating radiological consequences or, affect any fission product barriers. It does not increase any challenges to safety systems as well. Any changes to the information relocated to the TS Bases would be controlled under the TS Bases Control program, TS 5.5.14, which utilizes the criteria of 10 CFR 50.59 to determine if prior NRC [Nuclear Regulatory Commission] approval is required for any changes. Therefore, this proposed amendment [for each unit] would not significantly increase the consequences of an accident previously evaluated.

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

This proposed administrative change does not involve any changes to the design, operation, or maintenance of any structures[,] systems or components. The requirements in TS 3.5.5 for RWT operability will not be changed. This proposed amendment [for each unit] does not alter, degrade, or prevent actions described or assumed in an accident described in the PVNGS UFSAR from being performed. Any changes to the information relocated to the TS Bases would be controlled under the TS Bases Control program, TS 5.5.14, which utilizes the criteria of 10 CFR 50.59 to determine if prior NRC approval is required for any changes.

Therefore, the proposed amendment [for each unit] does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed administrative change does not involve any changes to the design, operation, or maintenance of any structures[,] systems or components. The requirements in TS 3.5.5 for RWT operability will not be changed. This proposed amendment [for each unit] does not alter, degrade, or prevent actions described or assumed in an accident. Any changes to the information relocated to the TS Bases would be controlled under the TS Bases Control program, TS 5.5.14, which utilizes the criteria of 10 CFR 50.59 to determine if prior NRC approval is required for any changes. Therefore, the proposed change does not involve a significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three 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: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: June 6, 2000.

Description of amendments request: The proposed amendments would restrict the emergency diesel generator (DG) acceptance criteria for steady-state voltage and frequency in several surveillance requirements (SRs) involving DG starts in Technical Specification (TS) 3.8.1, ''AC Sources—Operating,'' of the TSs for the three units. The amendments would also add a note to each SR that states: ''The steady state voltage and frequency limits are analyzed values and have not been adjusted for instrument error.'' The restricted acceptance criteria is to ensure proper DG operation.

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

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

No. The proposed change does not significantly increase the probability of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The more restrictive steady-state voltage and frequency ranges ensure that the equipment being powered by the diesel generator will function as required to mitigate an accident as described in the UFSAR. The diesel generators are part of the systems required to mitigate an accident. Mitigation equipment is not a factor in accident initiation and, therefore, the probability of an accident previously evaluated will not be significantly increased.

The change to the steady state diesel generator voltage and frequency acceptance limits does not increase the probability of a diesel generator failure [or a failure of offsite power]. Therefore, this change does not increase the probability of a station blackout event.

The consequences of an accident previously evaluated in the UFSAR will not be significantly increased. The more restrictive change to the diesel generator steady-state voltage and frequency acceptance limits ensures that the equipment powered by the diesel generators will perform as analyzed and mitigate the consequences of any accident described in the UFSAR. Therefore, the change in steady-state voltage and frequency acceptance limits is within the bounds of previously analysis in the UFSAR and does not involve a significant increase in the consequences of an accidently previously evaluated.

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

No. The possibility of an accident of a new or different kind from any accident previously evaluated has not been created. The more restrictive change to the diesel generator steady-state voltage and frequency acceptance limits ensures that the equipment powered by the diesel generators will perform as analyzed. This equipment and the diesel generators mitigate the consequences of an accident. Mitigation equipment does not contribute to accident initiation. Making existing requirements more restrictive will not alter the plant configuration (no new or different type of equipment will be installed) or change the methods governing normal plant operation. These changes are consistent with the assumptions made in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The change to the diesel generator steady-state voltage and frequency acceptance limits ensures that the equipment powered by the diesel generators will perform as analyzed. This equipment and the diesel generators mitigate the consequences of an accident. This change maintains the required function of the equipment powered by the diesel generators and ensures the required operation of the plant and any structures[,] systems, or components is as intended by the safety analysis. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three 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: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 7, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) related to the Engineered Safety Features Actuation System (ESFAS) Instrumentation found in TS 3/4.3.1, TS 3/4.3.2, and the associated Bases. Specifically, the proposed change would revise surveillance test intervals and allowed outage times for ESFAS instrumentation in TS 3/4.3.2. The proposed revision is based on WCAP–10271, ''Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,'' its supplements, and the NRC approvals issued in the Safety Evaluation Reports (SERs) dated February 21, 1985, and February 22, 1989, and the Supplemental SER dated April 30, 1990. In addition, the licensee is proposing specific changes to the reactor trip system instrumentation in TS 3/4.3.1, which are directly associated with implementing the ESFAS relaxations proposed in the submittal.

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

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

The determination that the results of the proposed changes are within all acceptable criteria was established in the SERs prepared for WCAP–10271 Supplement 2 and WCAP–10271 Supplement 2, Revision 1 issued by letters dated February 22, 1989 and April 30, 1990. Implementation of the proposed changes is expected to result in an acceptable increase in total Engineered Safety Features Actuation System yearly unavailability. This increase, which is primarily due to less frequent surveillance, results in a small increase in core damage frequency (CDF) and public health risk. The values determined by the WOG [Westinghouse Owners Group] and presented in the WCAP for the increase in CDF were verified by Brookhaven National Laboratory (BNL) as part of an audit and sensitivity analyses for the NRC staff. Based on the small value of the increase compared to the range of uncertainty in the CDF, the increase is considered to be acceptable.

Removal of the requirement to perform the Reactor Trip System analog channel operational test on a staggered basis will have a negligible impact on the Reactor Trip System unavailability. Staggered Testing was initially imposed to address the concerns of common cause failures. HNP's [Harris Nuclear Plant's] program to evaluate failures for common cause, process parameter signal diversity, and normal operational test spacing yield most of the benefits of staggered testing.

The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes may affect the probability of failure of the RPS [reactor protection system], but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

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

The proposed changes do not involve hardware changes and do not result in a change in the manner in which the Reactor Protection System provides plant protection or the manner in which surveillances are performed to demonstrate operability. No change is being made which alters the functioning of the Reactor Protection System. Rather the likelihood or probability of the Reactor Protection System functioning properly is affected as described above.

Therefore the proposed changes do not create the possibility of a new or different kind of accident.

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

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of reduced testing, other than as addressed above, is to allow a longer time interval over which instrument uncertainties (*e.g.,* drift) may act. An evaluation has been performed to assure that the plant setpoints properly account for these instrument uncertainties over the larger time interval.

Implementation of the proposed changes is expected to result in an overall improvement in safety as follows:

a. Less frequent testing will result in fewer inadvertent reactor trips and inadvertent actuations of Engineered Safety Features Actuation System components.

b. Less frequent distraction of the operator and shift supervisor to attend to and support instrumentation testing will improve the effectiveness of the operating staff in monitoring and controlling plant operation.

The foregoing analysis demonstrates that the proposed amendment to HNP TS does not involve a significant increase in the probability or consequences of a previously evaluated accident, does not create the possibility of a new or different kind of accident, and does not involve a significant reduction in a margin of safety.

Based upon the preceding analysis, CP&L [Carolina Power & Light Company] concludes that the proposed amendment does not involve a significant hazards consideration.

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

Attorney for licensee: William D. Johnson, Vice President and Corporate

Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: May 11, 2000.

Description of amendment request: The proposed changes would revise Technical Specification Surveillance Requirement (SR) 3.6.1.3.8. SR 3.6.1.3.8 currently requires verification of the actuation capability of each excess flow check valve (EFCV) every 24 months. This proposed change would relax the SR frequency by allowing a ''representative sample'' of reactor instrument line EFCVs to be tested every 24 months, such that each reactor instrument line EFCV will be tested at least once every 10 years.

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

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

The current SR frequency requires each reactor instrument line EFCV to be tested every 24 months. The reactor instrument line EFCVs at WNP–2 are designed so that they will not close accidentally during normal operation, but will close if a rupture of the instrument line is indicated downstream of the valve, and have their status indicated in the control room. This proposed change allows a reduced number of reactor instrument line EFCVs to be tested every 24 months. There are no physical plant modifications associated with this change. Industry operating experience demonstrates a high reliability of these valves. Neither reactor instrument line EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore; there can be no increase in the probability of occurrence of an accident regarding this proposed change.

Reactor instrument lines connecting to the reactor coolant pressure boundary are equipped with EFCVs and also have a flow-restricting orifice inside containment and upstream of the EFCV. The consequences of an unisolable rupture of such an instrument line has been previously evaluated in WNP–2 FSAR 15.6.2. The instrument lines that penetrate primary containment conform to Regulatory Guide 1.11 (WNP–2 FSAR 7.1.2.4). Those instrument lines are Seismic Category I and terminate in instruments that are Seismic Category I (reference WNP–2 FSAR Table 6.2–16 note 27).

The sequence of events in WNP–2 FSAR Section 15.6.2.2 for a reactor instrument line break assumes a continuous discharge of reactor water through the instrument line until the reactor vessel is cooled and depressurized (5 hours). Although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by this previous evaluation. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the consequences of an accident previously evaluated as a result of this change.

The containment atmosphere and suppression pool instrument line EFCVs are required to remain open to sense containment atmosphere and suppression pool level conditions during postulated accidents. They are not required to close during an instrument line break assumed during normal plant operation nor is their design capable of closing during normal plant conditions. These EFCVs do not meet the criteria for inclusion in 10 CFR 50.36(c)(3) as they have no active safety function and thus relocation of their testing requirements to processes controlled under 10 CFR 50.59 cannot affect the probability or consequences of an accident previously evaluated.

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

This proposed change allows a reduced number of reactor instrument line EFCVs to be tested each operating cycle and that the testing requirements for containment atmosphere and suppression pool instrument line EFCVs be relocated to a process controlled under 10 CFR 50.59. No other changes in requirements are being proposed. Industry operating experience demonstrates the high reliability of these valves. The potential failure of a reactor instrument line EFCV to isolate by the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line rupture. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

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

The consequences of an unisolable rupture of an instrument line has been evaluated in WNP–2 FSAR Section 15.6.2 in accordance with the requirements of Regulatory Guide 1.11. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (5 hours). The only margin of safety applicable to this proposed change is considered to be that implied by this evaluation. Since a continuous discharge was assumed in this evaluation, any potential failure of a reactor instrument line EFCV to isolate as a result of reduced testing frequency is bounded by existing analysis and does not involve a significant reduction in the margin of safety.

There is no accident for which the containment atmosphere or suppression pool instrument line EFCVs are designed to actuate to the isolation position for mitigation. A postulated break of a containment atmosphere or suppression pool instrument line under normal operating conditions would not result in a condition that would create the ability for these EFCVs to operate because neither the containment pressure nor the suppression pool level head would be sufficient to result in their actuation. As these EFCVs have no active design or safety function, the relocation of testing requirements would not involve a significant reduction in the margin of safety. A postulated break of any instrument line simultaneous with a loss of coolant accident is beyond the design basis for the plant.

Based upon the above, the proposed amendment is judged to involve no significant hazards considerations.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 25, 2000.

Description of amendment request: The proposed amendment would change the action statements for Technical Specification (TS) 3.8.2.2, A.C. Distribution—Shutdown, and TS 3.8.2.4, D.C. Distribution—Shutdown, by replacing the requirement to establish containment integrity within 8 hours, with a requirement to immediately suspend core alterations, the movement of irradiated fuel assemblies, and any operations involving positive reactivity additions. Related changes to the associated Bases were also proposed.

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

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

The existing requirement to establish containment integrity upon a loss of a required AC or DC bus in Mode 5 or 6 is not relied upon in any ANO–2 [Arkansas Nuclear One, Unit 2] accident analysis. Other components that may be rendered inoperable upon the loss of a required AC or DC bus are governed by other TSs and associated action statements. Such functions include core cooling, reactor coolant makeup capabilities, the status of containment penetrations and openings, and reactor coolant inventory. The TSs that govern these functions provide appropriate actions to address the failure at hand. The proposed change[s] act to minimize the possibility of a fuel handling accident when a required AC or DC bus is inoperable by requiring the suspension of the handling of irradiated fuel and core alterations. In addition, ANO–2 has demonstrated that the offsite dose consequences of a fuel handling accident within the containment building remain well within 10 CFR 100 limits without taking credit for the containment's fission product control function. Deleting the requirement to establish containment integrity is not relevant to the initiation of any accident previously evaluated, nor does it significantly increase the consequences of any accident previously evaluated. Other TS LCOs [limiting conditions for operation] provide appropriate actions that address shutdown cooling (SDC), makeup capability and inventory, and other important functions. The proposed change deletes the requirement to establish containment integrity in favor of those actions that act to minimize the likelihood of a fuel handling accident or a positive reactivity excursion. The proposed change reduces unnecessary actions required upon the loss of an AC or DC bus and provide greater consistency with the philosophies of the RSTS [Revised Standard Technical Specifications].

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

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

The existing actions associated with shutdown mode AC and DC TS sources are not considered accident initiators. The proposed revision does not present a physical change to plant systems or equipment. Deleting the requirement to establish containment integrity in favor of actions that aid in minimizing the likelihood of a fuel handling accident or positive reactivity excursion does not result in any new or different kind of accident from any previously evaluated.

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

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

The existing requirement to establish containment integrity upon a loss of any required AC or DC bus in Modes 5 or 6 acts to limit offsite release consequences should an accident occur during the period of inoperability. The proposed change acts to address the source, that is, aids in minimizing the likelihood of a fuel handling accident or an undetected positive reactivity addition while in Modes 5 and 6. By suspending all handling of irradiated fuel and core alterations, the likelihood of a fuel handling accident occurring is minimized. Since the loss of a required AC or DC bus could impact plant instrumentation, the suspension of all activities involving positive reactivity additions aids in preventing the impact of a positive reactivity addition from being undetected. Other possible Mode 5 and 6 conditions (loss of inventory, loss of shutdown cooling, etc.) are addressed in other shutdown mode TSs. In addition, ANO–2 has demonstrated that the offsite dose consequences of a fuel handling accident within the containment building remain well within 10 CFR 100 limits without taking credit for the containment's fission product control function. Since the proposed change exchanges accident mitigation strategy in favor of accident prevention strategy, no significant reduction in the margin to safety is evident.

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of amendment request: May 1, 2000.

Description of amendment request: The proposed amendment would: (1) Revise Technical Specification (TS) requirements regarding the minimum number of radiation monitoring instrumentation channels required to be operable during movement of fuel within the containment; (2) revise the Modes in which the surveillance specified by Table 4.3–3, ''Radiation Monitoring Instrumentation Surveillance Requirements,'' Item 2.c.ii is required; (3) revise TS 3.9.4, ''Containment Building Penetrations,'' to allow both Personnel Air Lock (PAL) doors and other containment penetrations to be open during movement of fuel assemblies within containment, provided certain conditions are met; (4) revise applicability and action statement requirements of TS 3.9.4. to be for only during movement of fuel assemblies within containment; (5) revise periodicity and applicability of Surveillance Requirement (SR) 4.9.4.1; (6) revise SR 4.9.4.2 to verify flow rate of air to the supplemental leak collection and release system (SLCRS) rather than verifying the flow rate through the system; (7) add two new SRs, 4.9.4.3 and 4.9.4.4, for verification and demonstration of SLCRS operability; (8) modify TS 3/4.9.9 for the containment purge exhaust and isolation system to be applicable only during movement of fuel assemblies within containment; and, (9) revise associate TS Bases as well as make editorial and format changes.

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

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

The proposed amendment involves changes to accident mitigation system requirements. These systems are related to controlling the release of radioactivity to the environment and are not considered to be accident initiators to any previously analyzed accident.

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

Based on the current technical specification requirements, an environmental release due to a fuel handling accident (FHA) occurring within containment is precluded by a design which automatically isolates the containment following detection of radioactivity by redundant containment purge monitors. The proposed amendment, which permits containment penetrations to be open during movement of fuel assemblies within containment, increases the dose at the site boundary and the control room operator dose due to a FHA occurring within containment; however, the dose remains within acceptable limits. Based on a radiological analysis of a FHA within containment with open containment penetrations being filtered by the Supplemental Leak Collection and Release System (SLCRS), the resultant radiological consequences of this event are well within the 10 CFR Part 100.11 limits, as defined by acceptance criteria in the Standard Review Plan (SRP) Section 15.7.4. Control room operator doses remain less than the 10 CFR Part 50 Appendix A General Design Criteria (GDC) 19 limit of 5 rem whole body or its equivalent to any part of the body. The proposed changes to LCO 3.9.4 and associated surveillance requirements will ensure that SLCRS filtration assumptions in the associated radiological analysis are met.

LCO 3.9.10 titled ''Water Leve—Reactor Vessel'' will continue to ensure that at least 23 feet of water is maintained over the fuel during fuel movement when the plant is in Mode 6. LCO 3.9.3 titled ''Decay Time'' will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 150 hours after shutdown. These LCOs will continue to ensure that two of the key assumptions used in the radiological safety analysis are met.

The radiological consequences of the Core Alteration events other than the FHA remain unchanged. These events do not result in fuel cladding integrity damage. A radioactive release to the environment is not postulated since the activity is contained in the fuel rods. Therefore, the affected containment systems are not required to mitigate a radioactive release to the environment due to a Core Alteration event.

The proposed revision in the minimum number of the Containment Purge Exhaust Radiation Monitoring Instrumentation channels required to be operable from one to two, ensures that redundant instrument channels are available to detect and initiate isolation of the containment purge and exhaust containment penetrations during a FHA inside containment.

The proposed administrative, editorial, and format changes do not affect plant safety. The Bases section has been revised as necessary to reflect the changes to these Specifications. Bases Section 3/4.9.9 will also be revised to remove text pertaining to Mode 5 applicability that is not relevant to this specification.

Therefore, the proposed amendment does not significantly increase the consequences of any previously evaluated accident.

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

The proposed amendment affects a previously evaluated accident; e.g., FHA. The proposed amendment does not represent a significant change in the configuration or operation of the plant. The proposed amendment does not impact Technical Specification requirements for systems needed to prevent or mitigate other Core Alteration events. The filtered SLCRS that will be utilized to control and filter the radioactive release from a FHA occurring within containment is the same system (with the exception of the flow path to the filter banks) currently relied upon to control and filter the release from a FHA in the fuel building. The primary function of SLCRS is to ensure that radioactive leakage from the primary containment following a Design Basis Accident (DBA) or radioactive release due to a fuel building FHA is collected and filtered for iodine removal prior to discharge to the atmosphere at an elevated release point through a ventilation vent. This system will be relied upon to control the releases from open containment penetrations should a FHA occur inside of containment until such time that these open containment penetrations can be isolated. The proposed amendment contains the requirement to maintain the capability to close open containment penetrations within 30 minutes following a FHA inside containment.

The filtered SLCRS that will be relied upon to mitigate a FHA within containment is classified as Quality Assurance (QA) Category I, Safety Class 3 and Seismic Category I as stated in Updated Final Safety Analysis Report (UFSAR) Section 6.5.3.2.1 titled ''Design Bases.'' As described in UFSAR Section 6.5.1 titled ''Engineered Safety Feature Filter Systems,'' filtered SLCRS is considered to be an engineered safety features (ESF) filter system used to mitigate the consequences of accidents.

Therefore, the proposed amendment 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?*

Based on the current technical specification requirements, an environmental release due to a FHA occurring within containment is precluded by a design which automatically isolates the containment following detection of radioactivity by redundant containment purge monitors. The proposed amendment increases the dose at the site boundary and the control room operator dose due to a FHA occurring within containment; however, the dose remains within acceptable limits. The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced.

The revised radiological analysis based on the proposed amendment demonstrates that during a FHA inside containment, the projected offsite doses will be well within the applicable regulatory limits of 10 CFR Part 100.11 of 300 rem thyroid and 25 rem whole body, and are less than the more restrictive guidance criteria in the SRP Section 15.7.4 of 75 rem thyroid and 6 rem whole body. Control room operator doses are less than the 10 CFR Part 50 Appendix A GDC 19 limit of 5 rem whole body or its equivalent to any part of the body. This radiological analysis is based on all airborne activity reaching the containment atmosphere, as a result of a FHA inside containment, being released to the environment over a 2 hour period. The 2 hour release period is based on the guidance contained in Regulatory Guide 1.25 titled ''Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.'' The proposed amendment contains a Bases requirement to maintain the capability to close open containment penetrations within 30 minutes following a FHA inside containment. Completion of this action will reduce the dose consequence of a FHA within containment by terminating the release to the environment prior to all airborne activity being released from the containment.

The margin of safety for Core Alteration events other than the FHA is not significantly reduced due to this proposed amendment. The proposed amendment does not impact Technical Specification requirements for systems needed to prevent or mitigate such Core Alteration events. These events do not result in fuel cladding integrity damage. Therefore, a radioactive release to the environment is not postulated since the activity is contained in the fuel rods.

The proposed revision in the minimum number of the Containment Purge Exhaust Radiation Monitoring Instrumentation channels required to be operable from one to two, ensures that redundant instrument channels are available to detect and initiate isolation of the containment purge and exhaust containment penetrations during a FHA occurring inside containment.

The proposed changes to SR 4.9.4.1 and SR 4.9.9, to remove unnecessary detail on when these surveillances are required to be performed, are administrative in nature and do not affect plant safety.

The proposed revision of the words ''through the'' to the words ''to filtered'' in SR 4.9.4.2.a does not change the LCO 3.9.4 requirements. This change makes the LCO and surveillance requirements consistent. This change is administrative in nature and does not affect plant safety.

The proposed administrative, editorial, and format changes do not affect plant safety. The Bases section has been revised as necessary to reflect the changes to these Specifications. Bases Section 3/4.9.9 will also be revised to remove text pertaining to Mode 5 applicability that is not relevant to this specification.

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

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

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

NRC Acting Section Chief: Marsha Gamberoni.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit No. 3, Citrus County, Florida

Date of amendment request: May 31, 2000.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 Improved Technical Specifications (ITS) to add an additional Condition and Required Action to ITS 3.3.11, ''Emergency Feedwater Initiation and Control (EFIC) System Instrumentation.'' The Action would require tripping the affected reactor coolant pump (RCP) status signals to each of the four EFIC channels when one or more RCP status signals or Reactor Coolant Pump Power Monitors (RCPPMs) for up to two RCPs become inoperable. This action is intended to ensure continued operability of the EFIC RCP status function when one or more RCPPMs or their associated RCP status signals are inoperable. The amendment also proposes changes to ITS Table 3.3.11–1 to properly characterize the configuration of the signals from the RCPPMs to EFIC, and to clarify source of the Loss of Main Feedwater Pump signals to EFIC. The proposed changes to Table 3.3.11–1 are intended to provide consistency between Table

3.3.11–1 and information provided in the ITS Bases for the EFIC System Instrumentation Specification.

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

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

The EFIC system is not an initiator of any design basis accident. The EFIC RCP status signal function is intended to ensure emergency feedwater is available to automatically raise levels in the once through steam generator (OTSG) to the natural circulation setpoint in the event of a loss of reactor coolant system (RCS) forced flow.

The proposed license amendment adds clarifying information to ITS Table 3.3.11–1, and an additional Required Action to ITS 3.3.11 that assures continued operability of the RCP status function of the EFIC system in the event one or more RCPPMs or their associated RCP status signals become inoperable. The design functions of the EFIC system and the initial conditions for accidents that require EFIC will not be affected by the change. Therefore, the change will not increase the probability or consequences of an accident previously evaluated.

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

The proposed amendment involves no changes to the design or operation of the EFIC system. The RCPPMs are part of the design of the Emergency Feedwater Initiation and Control (EFIC) System, and are assumed to function properly in the accident analysis. The proposed amendment will assure that the EFIC system performs as assumed in the safety analysis in the event of a loss of RCS forced flow. The proposed amendment change will not affect the other EFIC functions, and will not create any new plant configurations. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment adds additional actions to be taken in the event one or more RCPPMs or their associated RCP status signals become inoperable, and provides clarifying information regarding the sources and configuration of signals to EFIC. The proposed amendment ensures appropriate actions are taken to restore the operability of the EFIC RCP status function in the event that one or more RCP status signals to EFIC are lost. Thus, the proposed amendment will not result in a reduction in the margin of safety.

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

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit No. 3, Citrus County, Florida

Date of amendment request: June 1, 2000.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 (CR–3) Improved Technical Specifications 3.4.14 to extend the interval for calibration of the containment sump monitor from the current 18 months to 24 months. The monitor is used to detect and measure reactor coolant system (RCS) leakage by monitoring changes in the level of water in the containment sump. Extending the interval to 24 months would make it consistent with the current CR–3 operating cycle.

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

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

The containment sump monitor is not an initiator of any design basis accident. This monitor is used during normal plant operation to measure and to trend the rate of change of containment sump fluid level.

The containment sump monitor does not perform any safety function as part of mitigating the consequences of a design basis accident. Separate safety-related instrumentation is used to determine post-accident containment sump and containment flood levels and to satisfy the requirements of Regulatory Guide (RG) 1.97 for post-accident monitoring instrumentation. Additionally, the containment sump monitor does not have any associated safety system setpoint. The level switch in the instrument circuit is used only for automatic pumping of sump fluid using the two containment sump pumps.

A longer interval between calibrations may result in some increase in the amount of drift that the containment sump level monitor might experience between calibrations. The behavior of instrumentation, including considerations such as the amount of drift that the instrument might experience between calibrations, is not an accident precursor. Thus, changes to instrument maintenance such as intervals for performance of calibration, and the behavior of instruments including such considerations as the amount of drift, do not affect the probability of an accident. The probability of an accident previously evaluated is independent of the amount of drift that the containment sump level monitor might experience.

The containment sump monitor is used to detect RCS leakage during normal operation and does not have an accident mitigation function. Additionally, the ability of the instrument to detect small leaks will not be affected by extending the calibration interval.

Based on the above, increasing the interval between calibrations does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed license amendment involves no changes to the design or operation of the containment sump level monitor. Extending the interval between calibrations of the containment sump level monitor from 18 months to 24 months might result in greater drift of the monitor during the period of operation. However, the only function of the monitor is to detect changes and trends in the containment sump level during normal operation and the amount of drift that the monitor has experienced does not affect its ability to measure such changes and trends of the containment sump level. Furthermore, changes in the behavior of instrumentation, such as the amount of drift that the instrument might experience between calibrations, do not create the possibility of a new or different kind of accident.

Because initiation of accidents is independent of instrumentation behavior parameters such as drift, extending the calibration interval from 18 to 24 months does not create the possibility of any new or different kind of accident from any previously evaluated.

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

The CR–3 operating license, i.e., the Improved Technical Specifications, requires that instrumentation to detect leakage of reactor coolant system (RCS) inventory be available and operable during power operation. The required instrumentation is one containment sump monitor and one containment atmosphere radioactivity monitor.

The proposed extension of the containment sump monitor calibration interval from 18 to 24 months does not compromise the ability of the instrumentation to perform its safety function, *i.e.*, early detection of RCS leakage. This is so because the only function of the containment sump monitor is to detect changes and trends in the containment sump level during normal operation. The proposed license amendment makes no changes to either the design or operation of the sump monitor. The proposed license amendment makes no changes to the license requirements or to the design or operation of the containment atmosphere radioactivity monitor.

Because no changes are made to either the design or operation of the sump monitor, the sump monitor remains operable with the requested changes, and no changes are made to the containment atmosphere radioactivity monitor, FPC concludes that the change does not result in a significant reduction in a margin of safety.

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

Attorney for licensee: R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC–A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Richard P. Correia.

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: June 8, 2000.

Description of amendment requests: The proposed amendments would approve changes to the Updated Final Safety Analysis Report (UFSAR) to allow the use of probabilistic risk assessment (PRA) techniques in evaluating the need for tornado-generated missile barriers.

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

(1) *Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated*?

The possibility of a tornado reaching the Donald C. Cook Nuclear Plant (CNP) site is a design basis event considered in the UFSAR. The proposed change does not affect the probability that a tornado will reach the CNP site. However, the change affects the probability assumed in the current licensing basis that missiles generated by the winds of a tornado might strike certain plant systems or components.

No other accident scenarios, new initiators, or event precursors are affected or introduced by this change. There are a limited number of safety-related components that could potentially be struck by a tornado-generated missile. The total (aggregate) probability of exceeding 10 CFR 100 guidelines resulting from tornado missile strikes remains below the acceptance criterion ensuring overall plant safety. Thus, the proposed change does not constitute a significant increase in the probability of occurrence of an accident.

This change does not result in an increase in the quantity of radioactive materials potentially available for release to the environment in the event of an accident. The principle barriers to the release of radioactive materials are not modified or affected by this change. No new release pathways are created. Thus, the proposed change does not significantly affect potential offsite dose consequences.

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 possibility of a tornado reaching CNP site is a design basis event considered in the UFSAR. This change recognizes the acceptability of performing tornado missile probability calculations in accordance with established regulatory guidance. The change, therefore, deals with an established design basis event (the tornado). The change does not affect or create new accident initiators or precursors. Therefore, the change does not contribute to the possibility of a new or different kind of accident from those previously analyzed.

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

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

The existing licensing basis for CNP, with respect to the design basis event of a tornado reaching the plant, generating missiles, and directing them toward safety-related systems and components, is to provide positive missile protection for every required SSC [System, Structure, and Component] or area. This change recognizes the extremely low probability, below an established acceptance limit, that a limited subset of SSCs, and areas could be struck. This change from ''protecting all required systems, structures, and components'' to an ''extremely low probability of exceeding 10 CFR 100 guidelines as a result of tornado-generated missiles,'' does not constitute a significant decrease in the margin of safety due to the extremely low probability.

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.

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

Date of amendment requests: May 15, 2000.

Description of amendment requests: The proposed amendments would change Technical Specification 3.7.B.6 one time only to explicitly allow de-energizing Motor Control Center (MCC) 1T1 and MCC 1T2. The proposed change would allow either MCC 1T1 or MCC 1T2, one at a time, to be out of service for up to 72 hours provided the redundant MCC, its associated 480 Volt bus is verified operable, and the diesel generator and safeguards equipment associated with the redundant MCC are operable. The reason for the change is to install transfer switches for MCC 1T1 and MCC 1T2 for personnel protection, and to increase the allowed outage time for the MCC's to ensure sufficient time to install the transfer switches. This would prevent a dual unit shutdown to install each transfer switch under current Technical Specification 3.7.B.6.

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

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

The proposed changes do not involve any systems, structures or components whose failure would initiate an accident, thus, this change does not affect the probability of an accident.

The proposed changes extend the allowed out of service time for MCC 1T1 and MCC 1T2. The proposed changes would be applied only in support of a one-time modification to install transfer switches for the affected MCC's. The proposed changes do not extend the allowed out of service time for any components, supplied by these MCC's, that are relied on to mitigate the consequences of an accident. Thus, this change does not significantly increase the expected consequences of an accident.

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

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

The proposed changes do not change the way any systems, structures or components are operated. Nor does the proposed change introduce any new failure modes.

Therefore, the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed changes do not extend the allowed out of service times for any safety related components powered by the affected MCC's. Further, the proposed changes only allow one train (one of the affected MCC's) to be out of service and only if the opposite train MCC, its supporting sources and supplied safeguards equipment is verified operable. Thus, the proposed changes do not substantially impact the ability of operators to protect the fuel cladding, reactor coolant system or containment.

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

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: May 12, 2000.

Description of amendment requests: The proposed amendment would allow the design upgrade of the refueling water purification (RWP) system from design class II/non-seismic category 1 to design class I/seismic category 1 for purposes of permitting the cleanup of the refueling water storage tank (RWST) water while the RWST is required to be operable. This license amendment request (LAR) also proposes to allow the crediting of operator action to isolate a manual code boundary valve connected to the RWST following a seismic event or safety injection. It is desired to take suction from the RWST through an existing tank drain line to facilitate RWST recirculation through a non-seismically qualified reverse osmosis system while the RWST is required to be operable. This reverse osmosis system will be used to remove silica from the RWST water.

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

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

The upgrade of the refueling water purification (RWP) system piping will allow connection of the RWP system to the refueling water storage tank (RWST) while the RWST is required to be operable. The installation and use of a reverse osmosis (RO) system will allow removal of silica from the RWST while the RWST is required to be operable. The upgrade to the RWP system piping and use of the RO system does not involve any changes or create any new interfaces with the reactor coolant system or main steam system piping. Operation of the RWST is required to mitigate a loss-of-coolant and main steam line break accident, therefore, the connection of the RWP system to the RWST and use of the RO system would not affect the probability of these accidents occurring.

Neither the RWP system nor the RO system are credited for safe shutdown of the plant or accident mitigation. The upgrade to the RWP system piping to seismic category I will prevent seismically induced failure of the RWP system piping and thus prevent a loss of RWST inventory while the RWP system is connected to the RWST. The RWST can perform its safety function with an active failure in the RWP system in the short term phase of an accident while the RWP system is connected to the RWST. The RWST can perform its safety function with an active or passive failure in the RO system in the short term phase of an accident. Since the RWST inventory is not credited in the long term phase of an accident, active and passive failures in the RWP or RO system in the long term phase of an accident need not be considered.

Continuous operation of the RWP pump during a design basis event will not reduce the RWST water inventory nor the emergency core cooling system (ECCS) pump suction supply. The increase in RWST discharge flow due to an operating RWP pump will not adversely impact the required net positive suction head of the operating ECCS pumps.

A combination of design and administrative controls ensure that both the RWP and RO systems maintain RWST boron concentration and tank volume requirements whenever the contents of the RWST are processed through these systems. Potential boron dilution or volume losses of the RWST inventory during tank processing through the RWP system is prevented by administratively maintaining closed all manual boundary valves within the RWP system while the RWP system is used to clarify RWST contents. Prior to RO system operation, the RWST volume margin will be verified to be adequate to compensate for postulated RO system line losses and process losses through the RO system reject waste stream. The waste stream losses will be monitored throughout RO system operation. The RO system is designed to maintain a high boron recovery rate, which will be verified through testing prior to initial installation. Potential boron dilution during each batch operation of the RO system is prevented through verifying RWST boron margin prior to RO system operation and monitoring the RO system boron recovery rate by grab samples taken of the system inlet and outlet after each batch operation. Following each batch operation of the RO system, RWST mixing and sampling will be performed to verify the RWST boron concentration, and boron additions to the RWST will be made accordingly. Since the RWST will continue to perform its safety function, overall system performance is not affected, assumptions previously made in evaluating the consequences of the accident are not altered, and the consequences of the accident are not increased.

Therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

The upgrade of the RWP system piping to seismic category I will prevent seismically induced failure of the RWP piping. An active RWP pump failure will not result in a loss of the RWST safety function. An active or passive failure in the RO system will not result in loss of the RWST safety function. Adequate RWST volume and boron margin will be verified prior to RO system operation, the RO system boron recovery rate will be monitored by grab samples taken of the system inlet and outlet after each batch operation, a flow limiting device will limit the maximum potential RWST inventory loss rate to a low value, and operator action can be taken within 1 hour to isolate the RO system from the RWST. The upgrade to the RWP system and use of the RO system do not impact any other systems and thus cannot create a new failure mode in another system which could potentially create a new type of accident.

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

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

Neither the RWP system nor the RO systems are credited for safe shutdown of the plant or accident mitigation. The upgrade to the RWP system piping to seismic category I will prevent seismically induced failure of the RWP system piping and prevent loss of RWST inventory due to a seismic event while the RWP system is connected to the RWST. The RWST can perform its safety function with an active failure in the RWP system in the short term phase of an accident while the RWP system is connected to the RWST. The RWST can perform its safety function with an active or passive failure in the RO system in the short term phase of an accident. Since the RWST inventory is not credited in the long term phase of an accident, active and passive failures in the RWP or RO system in the long term need not be considered.

Adequate RWST volume and boron margin will be verified prior to RO system operation, a flow limiting device will limit the maximum inventory loss rate to a low value, and operator action can be taken within 1 hour to isolate the RO system from the RWST. The RO system waste stream losses will be monitored throughout RO system operation.

Potential boron dilution of the RWST inventory during tank processing through the RWP system is prevented by administratively maintaining closed all manual boundary valves within the RWP system while the RWP system is connected to the RWST. The

RO system is designed to maintain a high boron recovery rate, which will be verified through testing prior to initial installation. Potential boron dilution during each batch operation of the RO system is prevented through verifying RWST boron margin prior to RO system operation and monitoring the RO system boron recovery rate by grab samples taken of the system inlet and outlet after each batch operation. Following each batch operation of the RO system, RWST mixing and sampling will be performed to verify the RWST boron concentration, and boron additions to the RWST will be made accordingly. These measures will ensure the TS minimum RWST boron concentration is available to mitigate the short term consequences of a small break LOCA, large break LOCA, or main steam line break accident.

Therefore, the change does not involve a significant reduction in a margin of safety as defined in the basis for any technical specification.

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

NRC Section Chief: Stephen Dembek.

Pacific Gas and Electric Company, Docket No. 50–323, Diablo Canyon Nuclear Power Plant, Unit No. 2, San Luis Obispo County, California

Date of amendment requests: June 2, 2000.

Description of amendment requests: The proposed amendment would revise Technical Specification (TS) 3.5.2, ''ECCS—Operating,'' Action A, to change the allowed completion time for repair or replacement of the centrifugal charging pump (CCP) 2–1 during Cycle 10 of Unit 2 from 72 hours to 7 days. In response to high CCP 2–1 vibration, planning has been done for replacing the CCP 2–1 discharge head and bearing housing or to change out the entire CCP 2–1. The 72-hour allowed completion time is not sufficient to accomplish such emergent repairs on an inoperable CCP.

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

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

The emergency core cooling system (ECCS) and the centrifugal charging pumps (CCPs) are designed to respond to mitigate the consequences of an accident. They are not an accident initiator, and as such cannot increase the probability of an accident.

The loss of both CCPs, due to an inoperable CCP 2–1 and a single failure of CCP 2–2, could increase the consequences of an accident. A PRA was performed to evaluate the increased consequences. The worst case risk increment due to the increased completion time for CCP 2–1 and the maximum allowed out of service time is 2.5 percent. This is a non-significant risk increase for core damage frequency (CDF). Also, there is no noticeable increase in the large early release frequency as a result of this request.

Allowing 7 days to complete the repairs and post-maintenance testing of CCP 2–1 is acceptable since the ECCS system remains capable of performing its intended function of providing at least the minimum flow assumed in the accident analyses. During the extended maintenance and test period, appropriate compensatory measures will be implemented to restrict high risk activity. The consequences of accidents, which rely on the ECCS system, will not be significantly affected.

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

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

There are no new failure modes or mechanisms created due to plant operation for an extended period to perform repairs and post-maintenance testing of CCP 2–1. Extended operation with an inoperable CCP does not involve any modification in the operational limits or physical design of the systems. There are no new accident precursors generated due to the extended allowed completion time.

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

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

Plant operation for 7 days with an inoperable CCP 2–1 does not adversely affect the margin of safety. During the extended allowable completion time the ECCS system maintains the ability to perform its safety function of providing at least the minimum flow assumed in the accident analyses. During the extended maintenance and test period, appropriate compensatory measures will be implemented to restrict high risk activity.

Therefore, the change does not involve a significant reduction in a margin of safety as defined in the basis for any Technical Specification.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 15, 2000.

Description of amendment request: The proposed change is to LGS Unit 1 Technical Specifications (TSs) Figure 3.4.6.1–1, ''Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure,'' and associated changes to TS Bases Section 3/4.4.6. The proposed change revises the pressure-temperature (P–T) limits by revising the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel (RPV) of Unit 1 from 12 effective full power years (EFPY) to a maximum of 32 EFPY. The proposed change also eliminates the requirement to maintain reactor coolant system within a narrow temperature band less that 212 °F during pressure testing.

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

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

There are no physical changes to the plant being introduced by the proposed changes to the P–T curves. The proposed changes do not modify the reactor coolant pressure boundary, *i.e.,* there are no changes in operating pressure, materials or seismic loading. The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. The proposed P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, in conjunction with ASME Code Cases N–640 and N–588. The proposed P–T curves were established in compliance with the methodology used to calculate the predicted irradiation effects on vessel beltline materials. Usage of these procedures provides compliance with the intent of 10 CFR Part 50, Appendix G, and provides margins of safety that ensure that failure of the reactor vessel will not occur. The proposed P–T curves prohibit operational conditions in which brittle fracture of reactor vessel materials is possible. Consequently, the primary coolant pressure boundary integrity will be maintained. Therefore, the proposed 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.

The proposed changes to the P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and ASME B&PV Code, Section XI, Appendix G, in conjunction with ASME Code Cases N–640 and N–588. Compliance with the proposed P–T curves will ensure that conditions in which brittle fracture of primary coolant pressure boundary materials are possible will be avoided. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the P–T curves do not affect any activities or equipment, and are not assumed in any safety analysis to initiate any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes reflect an update of the P–T curves to extend the reactor pressure vessel operating limit to 32 Effective Power Years (EFPY). The revised curves are based on the latest ASME guidance. These proposed changes maintain the relative margin of safety commensurate with that which existed at the time that the ASME B&PV Code, Section XI, Appendix G, was approved in 1974. The revised pressure-temperature limits, although less restrictive than the current limits, were established in accordance with current regulations and the latest ASME Code information. Because operation will be within these limits, the reactor vessel materials will continue to behave in a non-brittle manner, thus preserving the original safety design bases. No plant safety limits, set points, or design parameters are adversely affected by the proposed TS changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: 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.

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

Date of amendment request: May 15, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) requirements to test the remaining diesel generators when (1) One of the two independent off-site power sources is inoperable as delineated in TS 3/4.8.1, Action a, and (2) a diesel generator is inoperable for other than preventative maintenance reasons as delineated in TS 3/4.8.1, Action b.

The proposed change also (1) Expands the diesel generator loading band for the monthly, six-month, and the two hour loaded pre-requisite requirement for the hot restart test in accordance with the guidance of Regulatory Guide 1.9, ''Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants,'' Rev. 3, 1993; and (2) corrects an administrative error in a note associated with TS 3.8.1.2.

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

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

The emergency diesel generator system is not an accident initiator. Eliminating the requirement to demonstrate that the operable diesel generators function properly, when there is no evidence that the inoperability of the affected diesel generator is the result of a potential common mode failure, will not increase the probability or the consequences of previously evaluated accidents, which rely upon emergency power supplies.

Eliminating the testing of the diesel generators whenever a single off-site power source is inoperable does not establish operability of the remaining off-site power source. Operability is determined by the performance of surveillance 4.8.1.1.1.1.a.

Elimination of unnecessary starts (challenges) to the diesel generators will result in increased equipment reliability and hence improved overall reliability for emergency onsite power supplies, as follows:

(A) Reduce the overall engine degradation resulting from wear and tear of testing and reduce the probability of failure due to engine degradation, and,

(B) Minimize the number of entries into an equipment configuration where a potential challenge to the safety function exists during the period of the tests.

Expanding the band from 2500–2600 KW to 2330–2600 KW to accommodate instrument inaccuracy does not change any design parameter. The diesel generator will still be fully loaded (90% to 100% of continuous rating) in accordance with Reg. Guide 1.9, Rev. 3, Section 2.2.2. The full capability of the diesel generator to carry its load will continue to be demonstrated during the 24 endurance run, which is unaffected by this request.

The proposed change to the note in TS 3.8.1.2 is a correction of an administrative oversight (renumbering of a surveillance requirement) and does not change the surveillance content or intent.

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

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

Eliminating the requirement to demonstrate that the operable diesel generators function properly affects testing requirements only and does not alter the physical configuration of the plant, replace or modify existing equipment, affect operating practices or create any new or different accident precursors.

Similarly, expanding the band from 2500–2600 KW to 2330–2600 KW to accommodate instrument inaccuracy does not change the manner in which the diesel generator is operated, or introduces any new or different failure from any previously evaluated.

The proposed change to the note in TS 3.8.1.2 is a correction of an administrative oversight (renumbering of a surveillance requirement) and does not change the surveillance content or intent.

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

3. *Does not involve a significant reduction in a margin of safety.*

Eliminating the testing of the diesel generators whenever a single off-site power source is inoperable does not establish operability of the remaining off-site power source. Operability of the remaining off-site power source is determined by the performance of surveillance 4.8.1.1.1.1.a. The normally performed monthly surveillance ensures the diesel will be available to perform their safety function.

Eliminating the requirement to demonstrate that the operable diesel generators function properly, when there is no evidence that the inoperability of the affected diesel generator is the result of a potential common mode failure, does not reduce the margin of safety. If the evaluation is inconclusive or determines that a cause of inoperability for a diesel generator is a potential common mode failure then operability testing will be conducted for the remaining operable diesels. This action will assure that the initial assumption of two independent power supplies, utilized in the accident analysis, remain valid.

The proposed changes do not adversely affect the ability of the diesels to operate when called upon. Rather, these changes should result in improved overall reliability of the diesels and therefore the margin of safety is preserved for those events in which there is a dependence upon on-site AC power supplies.

Expanding the band from 2500–2600 KW to 2330–2600 KW to accommodate instrument inaccuracy does not introduce any new or different failure from any previously evaluated or changes the manner in which the diesel generator is operated. Expanding the band does not change any instrumentation set point, or changes to the auto loading sequence of the diesel. The capability of the diesel to be loaded to its manufactured maximum ratings will continue to be demonstrated during the performance of the diesel endurance run, which is unaffected by this request.

The proposed change to the note in TS 3.8.1.2 is a correction of an administrative oversight (renumbering of a surveillance requirement) and does not change the surveillance content or intent.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 25, 2000.

Brief description of amendment: The proposed amendment would revise the Comanche Peak Steam Electric Station, Units 1 and 2, Technical Specifications, Limiting Condition for Operation (LCO) 3.9.4, ''Containment Penetrations,'' to allow certain containment penetrations to be open during refueling activities under appropriate administrative controls.

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?*

Response: No.

The proposed change to Technical Specification (TS) 3.9.4, ''Containment Penetrations,'' would allow certain containment penetration flow paths to be open during core alterations and movement of irradiated fuel within containment under specific administrative controls. The fuel handling accident [(FHA)] radiological analysis does not take credit for containment isolation or filtration. Therefore, the time required to close any open penetrations is not relevant to the confirmatory radiological analysis dose calculations and the proposed change does not involve a significant increase in the consequences of an accident previously evaluated. The proposed administrative controls for containment penetrations are conservative even though not required by the accident analysis.

The status of the penetration flow paths during refueling operations has no affect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Because the FHA outside containment remains the limiting accident and the probability of a accident is not affected by the status of the penetration flow paths, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The open containment penetration flow paths are not accident initiators and do not represent a significant change in the configuration of the plant. The proposed allowance to open the containment penetrations during refueling operations will not adversely affect plant safety functions or equipment operating practices such that a new or different accident could be created. Therefore, the proposed revision will not create a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Technical Specification LCO 3.9.4 closure requirements for containment penetrations ensure that the consequences of a postulated FHA inside containment during core alterations or fuel handling activities are minimized. The LCO establishes containment closure requirements, which limit the potential escape paths for fission products by ensuring that there is at least one integral barrier to the release of radioactive material. The proposed change to allow the containment penetration flow paths to be open during refueling operations under administrative controls does not significantly affect the expected dose consequences of a FHA because the limiting FHA is not changed. The proposed administrative controls provide assurance that prompt closure of the penetration flow paths will be accomplished in the event of a FHA inside containment thus minimizing the transmission of radioactive material from the containment to the outside environment. Under the proposed TS change, the provisions to promptly isolate open penetration flow paths provide assurance that the offsite dose consequences of a FHA inside containment will be minimized. Therefore, the proposed change to the Technical Specifications does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Robert A. Gramm

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

Date of amendment request: June 23, 2000.

Description of amendment request: The proposed amendment would revise Limiting Condition for Operation (LCO) 3.9.4, ''Containment Penetrations,'' of the technical specifications (TS) to allow certain containment penetrations to be open during refueling operations under administrative controls. The amendment would (1) Revise the note in the LCO for containment penetrations that may be open under administrative controls, deleting the reference to penetrations P–63 and P–98, and (2) delete the exception for penetrations P–63 and P–98 in Surveillance Requirement (SR) 3.9.4.1. In addition, there would be format and editorial corrections to TS 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Start Air,'' and TS 5.2.2.b, ''Administrative Controls,'' to remove errors in the conversion to improved TSs issued March 31, 1999, in Amendment No. 123. There are also changes to the TS Bases for the proposed changes to LCO 3.9.4 and SR 3.9.4.1.

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

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

The status of the penetration flow paths during refueling operations has no [effect] on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a FHA [fuel handling accident] inside containment with open penetration flow paths are bounded by the current analysis described in the USAR [updated safety analysis report for Wolf Creek] and the probability of an accident is not affected by the status of the penetration flow paths, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements in the technical specifications.

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

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

The open containment penetration flow paths are not accident initiators and do not represent a significant change in the configuration of the plant. The proposed allowance to open the containment penetrations during refueling operations will not adversely affect plant safety functions or equipment operating practices such that a new or different accident could be created.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements [in] the technical specifications.

Therefore, the proposed revision will not create a new or different kind of accident from any accident previously evaluated.

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

Technical Specification LCO 3.9.4 closure requirements for containment penetrations ensure that the consequences of a postulated FHA inside containment during core alterations or fuel handling activities are minimized. The LCO establishes containment closure requirements, which limit the potential escape paths for fission products by ensuring that there is at least one integral barrier to the release of radioactive material. The proposed change to allow the containment penetration flow paths to be open during refueling operations under administrative controls does not significantly affect the expected dose consequences of a FHA because the limiting FHA is not changed. The proposed administrative controls provide assurance that prompt closure of the penetration flow paths will be accomplished in the event of a FHA inside containment thus minimizing the transmission of radioactive material from the containment to the outside environment. Under the proposed TS change, the provisions to promptly isolate open penetration flow paths provide assurance that the offsite dose consequences of a FHA inside containment will be minimized.

The proposed changes to correct editorial/format errors involve corrections to the technical specifications that are associated with the original conversion application and supplements or the certified copy of the improved Technical Specifications. As such, these changes are considered as administrative changes and do not modify, add, delete, or relocate any technical requirements in the technical specifications.

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

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

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

Date of amendment request: May 22, 2000.

Description of amendment request: The proposed amendment would add new Technical Specifications (TSs) 3.7.2.a(ii) and 3.7.2.h to address voltage on the 230 kV (kilovolt) grid as a precondition of criticality and to provide a time limit for when the 230 kV grid voltage is found to be insufficient to support Loss-of-Coolant Accident (LOCA) electrical loading during power operation. The application also requests various minor editorial changes. The Bases have also been changed to reflect the addition of the two new TS and to provide clarification of the components to which surveillance is applicable.

*Date of publication of individual notice in **Federal Register:*** June 2, 2000 (65 FR 35404).

Expiration date of individual notice: July 3, 2000.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: May 4, 2000, as supplemented May 9, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 4.12.1.3, for the control building automatic isolation and recirculation dampers to remove the individual damper component tag numbers. The surveillance requirements do not change. The associated Bases is also changed to reflect the applicable section of the Updated Final Safety Analysis Report.

Date of issuance: June 29, 2000.

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

Amendment No.: 223.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 22, 2000 (65 FR 32132).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 19, 1999, as supplemented April 21, 2000.

Brief description of amendments: The amendments approved changes in the Updated Final Safety Analysis Report (UFSAR) that constitute an unreviewed safety question as described in 10 CFR 50.59. These changes increase the probability of occurrence of a malfunction. These changes were not previously evaluated in the UFSAR, specifically, Section 5.3.1, ''External Missiles'' of the UFSAR did not address the probability of a missile from Unit 1 turbine-generator striking: (1) The refueling water tanks, (2) the No. 11 fuel oil storage tank, and (3) the plant equipment through various roof slabs or through non-missile-proof openings in the missile-proofing walls. The UFSAR only discusses a turbine missile strikingthe containment, control room, switchgear room, and waste processing area. The amendment authorizes the licensee to revise the turbine missile analysis to include the additional targets.

Date of issuance: June 19, 2000.

Effective date: As of the date of issuance to be implemented by December 31, 2000.

Amendment Nos.: 236 and 210.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register:** December 15, 1999 (64 FR 70079).

The April 21, 2000, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated June 19, 2000.

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Inc., Docket Nos. 50–003 and 50–247, Indian Point Nuclear Generating Station, Units 1 and 2, Buchanan, New York.

Date of amendment request: February 14, 2000.

Brief description of amendments: The amendments would eliminate from Environmental Technical Specifications Section 5.4.1, Routine Reports, the discussion regarding Section 4.2. Specifically, the proposed change seeks to delete the reference to and discussion about Section 4.2, which was deleted as part of Amendment No. 90 to Operating License No. DPR–26.

Date of issuance: June 8, 2000.

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

Amendment Nos.: 47 to DPR–5, and 210 to DPR–26.

Facility Operating License Nos. DPR–5 and DPR–26: The amendments revised the Environmental Technical Specifications.

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17912). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application of amendments: April 13, 2000, as supplemented by letter dated May 30, 2000.

Brief description of amendments: The amendments revise the Technical Specifications and associated Bases pages to accommodate the use of Mark-B11 fuel with M5 cladding.

Date of Issuance: June 21, 2000.

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

Amendment Nos.: 313, 313, and 313.

Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 17, 2000 (65 FR 31356).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 29, 1999.

Brief description of amendment: The amendment revised Surveillance Requirement 3.5.2.2. The change requires maintaining a higher level in the condensate storage tanks.

Date of issuance: June 20, 2000.

Effective date: June 20, 2000, to be implemented within 30 days from the date of issuance.

Amendment No.: 165.

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

Date of initial notice in **Federal Register:** August 25, 1999 (64 FR 46431).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 20, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 1, 1999, as supplemented by letter dated May 10, 2000.

Brief description of amendment: This amendment revised the frequency of performing Technical Specification Surveillance Requirement (SR) 3.6.1.7.4, verification that each containment spray nozzle is unobstructed. The frequency for performing SR 3.6.1.7.4 has been changed from once every 10 years to conditions following maintenance which could result in nozzle blockage.

Date of issuance: June 29, 2000.

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

Amendment No.: 113.

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

Date of initial notice in ***Federal Register:*** December 15, 1999 (64 FR 70088).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 2000.

No significant hazards consideration comments received: No.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: July 7, 1999.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to change the component surveillance frequencies for the following TSs to indicate a frequency of once per 3 months: Core Spray System TS 4.4.A.1 and 4.4.A.2, Containment Cooling System TS 4.4.C.1, Emergency Service Water System TS 4.4.D.1, Fire Protection System TS 4.4.F (isolation valves only), and Pressure Suppression Chamber—Drywell Vacuum Breakers TS 4.5.F.5.a. The TSs currently stipulate a component surveillance frequency of once per month. Also, the amendment revised TS pages 4.4–1 and 4.4–2 to incorporate editorial format changes and TS page 4.4–3 to accommodate the expanded text.

Date of Issuance: June 26, 2000.

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

Amendment No.: 210.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications. Date of initial notice in ***Federal Register:*** October 20, 1999 (64 FR 56531).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 26, 2000.

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: November 29, 1999.

Description of amendment request: The amendment relocates Surveillance Requirement 4.8.1.1.2f.1 which requires inspection of the Emergency Diesel Generator (EDGs) at least once per 18 months in accordance with procedures prepared in conjunction with its manufacturer from the Technical Specifications to the Seabrook Station Technical Requirements Manual.

Date of issuance: June 16, 2000.

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

Amendment No.: 71.

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

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4281).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 16, 2000.

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: November 30, 1999, as supplemented on April 28, 2000.

Description of amendment request: The amendment revises the Technical Specifications (TSs) by: (1) Inclusion of a new Administrative Control TS 6.7.6i for establishing, implementing, and maintaining a Diesel Fuel Oil Testing Program for testing new and stored fuel oil; (2) relocation of current surveillance requirement (SR) 4.8.1.1.2d and SR 4.8.1.1.2e.1, containing SRs for fuel oil sampling and testing, to the Diesel Fuel Oil Testing Program in the Seabrook Station Technical Requirements (SSTR) Manual; (3) revision of SR 4.8.1.1.2d to reference the Diesel Fuel Oil Testing Program as a surveillance requirement; (4) inclusion of additional surveillance requirements to SR 4.8.1.2 for checking and removing accumulated water from the day and storage fuel oil tanks, verifying new and stored fuel oil properties and visually inspecting diesel generator exhaust leakage when the plant remains in Modes 5 and 6 of operation; (5) relocation to the Diesel Fuel Oil Testing Program SR 4.8.1.12h for cleaning diesel fuel storage tanks at a 10-year frequency to the SSTR Manual; and (6) revision of TS Bases 3/4.8.1 by adding a statement that the exceptions to the certain Regulatory Guides are specified in the plant's Updated Final Safety Analysis Report.

Date of issuance: June 27, 2000.

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

Amendment No.: 73.

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

Date of initial notice in ***Federal Register:*** May 17, 2000 (65 FR 31358).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 27, 2000.

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: April 14, 2000.

Description of amendment request: This amendment revises the Technical Specifications by relocating Sections 3/4.9.5, "Communications", 3/4.9.6, "Refueling Machine", and 3/4.9.7, "Crane Travel—Spent Fuel Storage Areas" to the Seabrook Station Technical Requirement Manual.

Date of issuance: June 23, 2000.

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

Amendment No.: 72.

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

Date of initial notice in ***Federal Register:*** May 17, 2000 (65 FR 31358).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 23, 2000.

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: March 16, 2000, as supplemented by letters dated April 11, April 19, June 2, and June 9, 2000.

Brief description of amendments: The amendments revise several sections of the improved Technical Specification (ITS) to correct 19 editorial errors made in either (1) the application dated June 2, 1997, (and supplemental letters) for the ITSs, or (2) the certified copy of the ITSs that was submitted in the licensee's letters of May 19 and 27, 1999. The proposed amendment would also revise 10 instances of incorrect incorporation of the CTS into the ITS. One of the proposed editorial errors and one of the incorrect incorporations of the CTS will be addressed in a future letter. The ITSs were issued as License Amendments 135 and 135 dated May 28, 1999.

Date of issuance: June 21, 2000.

Effective date: June 21, 2000, to be implemented by June 30, 2000.

Amendment Nos.: Unit 1–142; Unit 2–142

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

*Date of initial notice in **Federal Register:*** April 19, 2000 (65 FR 21032).

The April 19, June 2, and June 9, 2000, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 27, 1999, as supplemented April 11, 2000.

Brief description of amendment: This amendment revises Technical Specifications (TSs) 4.6.2.2.b, ''Suppression Pool Spray,'' and 4.6.2.3.b, ''Suppression Pool Cooling,'' to modify the acceptance criteria associated with flow rate testing of the Residual Heat Removal system pumps.

Date of issuance: June 16, 2000.

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

Amendment No.: 128.

Facility Operating License No. NPF–57: This amendment revised the TSs.

*Date of initial notice in **Federal Register:*** January 26, 2000 (65 FR 4289)

The April 11, 2000, supplement 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 amendment is contained in a Safety Evaluation dated June 16, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 5, 1999, as supplemented December 3, 1999.

Brief description of amendment: The amendment revises the applicability for the reactor power distribution limits and Average Power Range Monitor gain adjustments. The applicability is revised to operation at ≥ 25% Rated Thermal Power.

Date of Issuance: June 21, 2000.

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

Amendment No.: 188.

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

*Date of initial notice in **Federal Register:*** December 29, 1999 (64 FR 73102)

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated June 21, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 19, 2000.

Brief description of amendments: These amendments revise Technical Specification 15.4.4–II.A to clarify that a different primary containment tendon may be designated a control tendon providing that the new control tendon has not previously been physically changed (*e.g.*, retensioned).

Date of issuance: June 27, 2000.

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

Amendment Nos.: 196 and 201.

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

*Date of initial notice in **Federal Register:*** March 8, 2000 (65 FR 12295).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 27, 2000.

No significant hazards consideration comments received: No.

Wisconsin Public Service Corporation, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: October 27, 1998, as supplemented on February 23, 2000.

Brief description of amendment: The amendment revises the plugging limits specified in TS 4.2.b, ''Steam Generator Tubes,'' for the Westinghouse hybrid-expansion-joint sleeve and the Westinghouse laser-welded sleeve. The proposed amendment also revises the list of applicable references specified in TS 4.2.b.

Date of issuance: June 27, 2000.

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

Amendment No.: 148.

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

*Date of initial notice in **Federal Register:*** November 18, 1998 (63 FR 64126). The February 23, 2000, supplement is within the scope of the original notice and does not change the proposed no significant hazards consideration finding.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 27, 2000.

No significant hazards consideration comments received: No.

Yankee Atomic Electric Company, Docket No. 50–29, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of application for amendment: March 17, 1999, as supplemented April 23, July 21, and November 2, 1999, and March 6, 2000.

Brief description of amendment: The amendment revises Technical Specification Section 6.0, Administrative Controls, by consolidating management positions and modifying review and audit functions.

Date of issuance: June 20, 2000.

Effective date: June 20, 2000.

Amendment No.: 154.

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

*Date of initial notice in **Federal Register:*** April 7, 1999 (64 FR 17033) The April 23, July 21, and November 2, 1999, and March 6, 2000, letters provided additional clarifying information that 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 amendment is contained in a Safety Evaluation dated June 20, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 5th day of July 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–17625 Filed 7–11–00; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 24553; 812–11908]

The Pitcairn Trust Company, et al.; Notice of Application

July 6, 2000.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of an application under sections 6(c) and 17(b) of the Investment Company Act of 1940 (the ''Act'') for an exemption from section 17(a) of the Act.

SUMMARY OF THE APPLICATION: Applicants request an order to permit certain common and collective trust funds, certain individual trust accounts and certain limited partnerships to transfer their assets to certain series of a registered open-end management investment company in exchange for shares of the series.

APPLICANTS: The Pitcairn Trust Company (''PTC''), Pitcairn Funds (the ''Trust''), Diversified Value Fund, Diversified Growth Fund, Select Value Fund, Select Growth Fund, Small Cap Value Fund, Small Cap Growth Fund, Tax Exempt Bond Fund, Family Heritage Fund and International Equity Fund (collectively, the ''Common Trust Funds''), Employee Benefit Large-Capitalization Fund, Employee Benefit Mid-Capitalization Fund, Employee Benefit Small-Capitalization Fund, Employee Benefit Fixed Income Fund and Employee Benefit International Equity Fund (collectively, the ''Collective Trust Funds,'' together with the Common Trust Funds, the ''CTFs''). Collectively, PTC, the Trust and the CTFs are referred to as ''Applicants.''

FILING DATES: The application was filed on December 23, 1999 and amended on June 29, 2000. Applicants have agreed to file an 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 issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving Applicants with a copy of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on July 27, 2000, and should be accompanied by proof of service on Applicants in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the Commission's Secretary.

ADDRESSES: Secretary, Commission, 450 Fifth Street, NW., Washington, DC 20549–0609. Applicants, c/o One Pitcairn Place, 165 Township Line Road, Suite 3000, Jenkintown, PA 19046.

FOR FURTHER INFORMATION CONTACT: Emerson S. Davis, Sr., Senior Counsel, at (202) 942–0714, or Janet M. Grossnickle, Branch Chief, at (202) 942–0564 (Division of Investment Management, Office of Investment Company Regulation).

SUPPLEMENTARY INFORMATION: The following is a summary of the application. The complete application may be obtained for a fee from the Commission's Public Reference Branch, 450 Fifth Street, NW., Washington, DC 20549–0102 (telephone (202) 942–8090).

Applicants' Representations

1. The Trust, a Delaware trust, will be registered under the Act as an open-end management investment company and will offer a number of series (each a ''Fund'') to the public, each with separate investment objectives, policies, and restrictions. PTC will serve as investment adviser to each Fund.[1]

2. PTC is a wholly-owned subsidiary of Pitcairn Company, which is wholly-owned by Pitcairn Group L.P. (''PGLP''), a limited partnership. PGLP's limited partnership units are owned by approximately 85 adult Pitcairn family members and related trusts, trusts governed by the Uniform Transfers to Minors Act, foundations and religious organizations supported by the Pitcairn family. For some of these family members, the beneficial ownership interests in PGLP partnership units are in excess of 5% of total units outstanding, both in terms of economic interest and voting power. Pitcairn family members also beneficially own, primarily through trusts, approximately 63% of the interests in the Common Trust Funds. A number of Pitcairn family members serve as co-trustees for trusts with, in the aggregate, more than 5% of the total beneficial interests in one or more common Trust Funds. Certain employee benefit plans maintained for the benefit of employees of PTC and its affiliates, including three members of the Pitcairn family who are employees of PTC (''PTC Plans''), own 70%–85% of the assets of the Collective Trust Funds.

3. Each CFT is maintained by PTC and is either (i) a ''common trust fund'' as defined in section 584(a) of the Internal Revenue Code of 1986, as amended (''Code''), or (ii) a collective trust fund that meets the requirements of section 401 of the Code. The CTFs are excluded from the definition of ''investment company'' under sections 3(c)(3) (for the Common Trust Funds) and 3(c)(11) (for the Collective Trust Funds) of the Act. Participants in the CFTs are persons or entities for which PTC acts as either trustee, executor, administrator, guardian, or custodian (''Participants''). Pitcairn company serves as the general partner of certain limited partnerships (''Partnerships''), the units of which are beneficially owned by clients of PTC. PTC serves as trustee for certain individual trust accounts (''ITAs'') that are held by PTC as sole or co-trustee for the benefit of individual clients, none of which is a Pitcairn family member or an entity in which a Pitcairn family member has a pecuniary interest.

4. Applicants propose to transfer in-kind all of the assets of each CFT, ITA and Partnership to one of the Funds with generally similar investment objectives in exchange for Class I shares of the respective Fund having an aggregate net asset value equal to that of the assets transferred (the ''Conversions'').[2] Class I shares will not be subject to a front-end or contingent deferred sales charge, redemption fees or rule 12b–1 distribution fees, although there may be a shareholder service fee of 0.25%. The assets of the CFTs to be transferred will be valued in accordance with the provisions of rule 17a–7(b) under the Act, and the shares of the Funds issued will have an aggregate net asset value equal to the value of the assets transferred. The shares of the

[1] As a ''bank'' within the meaning of section 202(a)(2) of the Investment Advisers Act of 1940 (''Advisers Act''), PTC currently is not subject to the registration requirements of the Advisers Act.

[2] Applicants also request relief for future transactions in which the assets of a terminating common or collective trust fund maintained by PTC are exchanged for shares of a registered open-end management investment company, or a series thereof, advised by PTC, or any entity controlling, controlled by, or under common control with PTC when owners of PTC or the PTC Plans own 5% or more of such trust fund or such registered investment company, or series thereof (''Future Transactions''). Applicants state that they will rely on the requested relief for Future Transactions only in accordance with the terms and conditions contained in the application.