Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Mr. James E. Lyons (telephone 301–415–7371), between 7:30 a.m. and 4:l5 p.m., EST.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at *http://www.nrc.gov/ACRSACNW.*

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., EST, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: December 20, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–33010 Filed 12–26–00; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 4, 2000, through December 15, 2000. The last biweekly notice was published on December 13, 2000.

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

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

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

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

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

By January 26, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, 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).

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: September 29, 2000.

Description of amendment request: The proposed amendments would make various changes to the Technical Specifications (TS) to support a change in fuel vendors from Siemens Power Corporation to General Electric and a transition to the use of GE 14 fuel.

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

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

Evaluation of effect on the probability of an accident previously evaluated:

1. *Administrative Changes.* The revisions to Current Technical Specifications (CTS) Sections 2.1.B, ''Thermal Power, High Pressure and High Flow,'' and 3.6.A, ''Recirculation Loops,'' regarding the Minimum Critical Power Ratio (MCPR) Safety Limit, the changes to CTS Section 6.9.A.6.b, ''Core Operating Limits Report,'' and the changes to the definitions are administrative changes and will not affect the probability of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology.* The changes to CTS Sections 3/4.3.C, ''Control Rod Operability,'' 3/4.3.D, ''Maximum Scram Insertion Times,'' 3/4.3.E, ''Average Scram Insertion Times,'' 3/4.3.F, ''Group Scram Insertion Times,'' 3/4.3.G, ''Control Rod Scram Accumulators,'' 3/4.3.H, ''Control Rod Coupling,'' and 3/4.3.I, ''Control Rod Position Indication System,'' revise the methodology for determining rod operability and control rod scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not affect the probability of an accident previously evaluated.

3. *Control Rod Scram Times.* The addition of required scram times for General Electric (GE) analyzed cores does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

4. *Rod Worth Minimizer (RWM).* The revision to CTS Section 3/4.3.L, ''Rod Worth Minimizer,'' lowers the power level at which the analyzed rod position sequence must be followed. This change does not affect plant systems, structures, or components. Because there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gram fuel design limit, the probability of an accident is not increased.

5. *Transient Linear Heat Generation Rate (TLHGR).* The revisions to CTS Section 3.11.B, ''*Transient Linear Heat Generation Rate*,'' add fuel thermal limits that are monitored to ensure that TLHGR is not violated. These changes do not physically alter plant systems, structures or components and therefore do not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. *Administrative Changes.* The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b regarding the COLR, and the changes to

the definitions are administrative changes and will not affect the consequences of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology.* The revisions to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I are made to ensure that appropriate scram times are reflected in the TS for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and local scram times (*i.e.,* a four control rod group). The proposed revisions, based on the Improved Technical Specification (ITS) methods, ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

In addition, numerous changes to the control rod operability and scram timing requirements were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.

Administrative changes. These consist of restructuring, interpretation, rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the consequences of an accident previously evaluated.

More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

Less restrictive changes. The less restrictive changes involve increasing the time to complete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies are not assumed in the analysis of the consequences of any accidents previously evaluated, and therefore, cannot increase the consequences of such accidents. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. The revised scram timing methods will result in operating thermal limits that will maintain the identical safety limits. Thus, the consequences of the evaluated accidents will not increase.

3. *Control Rod Scram Times.* Cycle-specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific events in the Updated Final Safety Analysis Report (UFSAR), GE has determined that there is negligible impact on results of events which are not analyzed on a cycle-specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

4. *RWM.* The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the Control Rod Drop Accident (CRDA) analysis are not violated. Compliance with the analyzed rod position sequence, and operability of the RWM is required in Mode 1, ''Power Operation,'' and Mode 2, ''Startup,'' when thermal power is less than or equal to 10% Rated Thermal Power (RTP). When thermal power is greater than 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA. Because the fuel design limit of 280 cal/gm is not exceeded, this change to lower the Low Power Setpoint (LPSP) does not increase the consequences of an accident previously evaluated.

5. *TLHGR.* The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure that the TLHGR limit is not violated. The TLHGR protects the fuel from 1% plastic strain and fuel centerline melt. Because these criteria have not changed, the consequences of an accident have not changed.

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

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

1. *Administrative Changes.* The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b regarding the COLR, and the changes to the definitions are administrative changes and will not create the possibility of a new or different kind of accident. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology.* The changes to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I revise the control rod operability and scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.

3. *Control Rod Scram Times.* These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.

4. *RWM.* The revisions to CTS Section 3/4.3.L lower the power level at which the analyzed rod position sequence must be followed. This change does not affect plant systems, structures, or components. Because there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit, no new or different type of accident is created.

5. *TLHGR.* The revisions to CTS Section 3.11.B revise the description of fuel thermal limits that are monitored to ensure that TLHGR is not violated. These changes are analytical in nature and do not affect plant systems, structures, or components. Therefore, the changes do not create the possibility of a new or different kind of accident.

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

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

1. *Administrative Changes.* The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the changes to CTS Section 6.9.A.6.b, regarding the COLR, and the changes to the definitions are administrative changes and will not reduce the margin of safety. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology.* The revisions to the CTS control rod operability and scram insertion times

ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and the local scram times (*i.e.,* a four control rod group). ITS methods ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, the changes do not involve a reduction in the margin of safety.

In addition, numerous changes to the control rod operability and scram timing requirements were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.

Administrative Changes. These consist of restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the margin of safety.

More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not reduce the margin of safety.

Less restrictive changes. The less restrictive changes involve increasing the time to complete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies have been extended for several reasons, including experience showing low probability of failures, the benefit of allowing time to perform actions without undue haste, or due to compensating changes in other actions. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. Thus, there is no significant reduction in the margin of safety.

3. *Control Rod Scram Times.* The addition of required scram times for GE analyzed cores based on GE analysis methodology does not involve a reduction in the margin of safety. For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure the MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

4. *RWM.* The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. Compliance with the analyzed rod position sequence, and operability of the RWM is required in Modes 1 and 2 when thermal power is less than or equal to 10% rated thermal power (RTP). When thermal power is greater than 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel design limit during a CRDA. Because the fuel design limit of 280 cal/gm is not exceeded above 10% RTP, this change to reduce the LPSP does not reduce a margin of safety.

5. *TLHGR.* The addition of the ratio of Maximum Fraction of Limiting Power Density (MFLPD) to the Fraction of Rated Thermal Power (FRTP) provides thermal limit protection for GE fuel. This provides equivalent protection to ensure that the TLHGR limit is maintained. Therefore, the revisions to CTS Section 3.11.B will not reduce a margin of safety.

Therefore, these proposed changes to the CTS do not involve a significant reduction in the margin safety.

Proposed Changes to ITS

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

Evaluation of the effect on the probability of an accident previously evaluated.

1. *Administrative Changes.* The revision to Improved Technical Specification (ITS) Section 5.6.5, ''Core Operating Limits Report,'' and the added definitions are purely administrative changes and do not affect the probability or consequences of an accident previously evaluated.

2. *Control Rod Scram Times.* The revision to ITS Table 3.1.4–1, ''Control Rod Scram Times,'' adds scram time requirements for GE analyzed cores. This change does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

3. *Average Power Range Monitor (APRM) Gain and Setpoint.* The revisions to ITS Section 3.2.4, ''Average Power Range Monitor (APRM) Gain and Setpoint,'' revise the description of fuel thermal limits that are monitored to ensure the TLHGR is not violated. The changes to this section are analytical in nature and do not affect plant systems, structures, or components and therefore will not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. *Administrative Changes.* The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and do not affect the probability or consequences of an accident previously evaluated.

2. *Control rod scram times.* The revisions to ITS Section 3.1.4, ''Control Rod Scram Insertion Times,'' are made to ensure the appropriate scram times are reflected in the Technical Specifications (TS) for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. Cycle specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific UFSAR events, GE has determined that there is negligible impact on the results of events which are not analyzed on a cycle specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient due to the TS changes.

3. *APRM Gain and Setpoint.* The revisions to ITS Section 3.2.4 will not increase the consequences of an accident previously evaluated. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated. The TLHGR protects the fuel from 1% plastic strain and fuel centerline melt. Because these criteria have not changed, the consequences of an accident have not changed.

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

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

1. *Administrative Changes.* The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and therefore do not create the possibility of a new or different kind of accident.

2. *Control Rod Scram Insertion Times.* The revisions to ITS Section 3.1.4 do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes to these sections revise the control rod scram time requirements for

operation. This change does not physically alter plant systems, structures, or components.

3. *APRM Gain and Setpoint*. The revisions to ITS Section 3.2.4 will not create the possibility of a new or different kind of accident. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The changes in this section revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated.

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

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

1. *Administrative Changes*. The revision to ITS Section 5.6.5 and the added definitions are purely administrative changes and do not affect the margin of safety.

2. *Control Rod Scram Insertion Times*. For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure that the MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

3. *APRM Gain and Setpoint*. The addition of MFLPD/FRTP provides thermal limit protection for GE fuel. This provides equivalent protection to ensure that the TLHGR limit is maintained. Therefore, the revisions to ITS Section 3.2.4 will not reduce a margin of safety.

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

Based on the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

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

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

NRC Section Chief: Anthony J. Mendiola.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: November 10, 2000.

Description of amendment request: The proposed amendments would revise several sections of the Technical Specifications (TS) and add a new TS section to incorporate Oscillation Power Range Monitor (OPRM) Instrumentation.

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

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

The proposed changes for LaSalle County Station will delete the thermal hydraulic instability administrative requirements and Power versus Flow figure and references to it from the TS, and insert a new TS for the OPRM instrumentation. The proposed TS will allow the enabling of the OPRM instrumentation trips. The deletion of the thermal hydraulic instability administrative requirements and Power versus Flow figure and the requirements to have an operable OPRM instrumentation trip does not have an effect on any accident previously evaluated or the associated accident assumptions. Thus, the proposed changes do not significantly increase the probability of an accident previously evaluated.

The proposed changes do not adversely affect the integrity of the fuel cladding, reactor coolant system or secondary containment. As such, the radiological consequences of previously evaluated accident are not changed. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

The proposed changes do not effect the assumed accident performance of any structure, system, or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

The OPRM instrumentation will initiate an automatic reactor trip upon detection of an instability that could threaten the Minimum Critical Power Ratio (MCPR) safety limit. The OPRM Instrumentation System consists of four (4) OPRM instrumentation trip channels. When one OPRM instrumentation module is inoperable, the remaining redundant OPRM Instrumentation module in the associated OPRM trip channel maintains the operability of the trip channel and thus there is no loss of trip function redundancy.

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

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

Boiling Water Reactors are susceptible to thermal hydraulic instabilities if operated at high power and low flow conditions. 10 CFR 50, Appendix A, General Design Criterion (GDC) 10, "Reactor design," requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Additionally, GDC 12, "Suppression of reactor power oscillation," requires the reactor core and associated coolant, control, and protection systems to be designed to assure that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed.

The detection and suppression of instability is required to insure that the MCPR safety limit is not exceeded during a transient. The OPRM instrumentation will initiate an automatic reactor trip upon detection of an instability that could threaten the MCPR safety limit.

The OPRM Instrumentation System consists of four (4) OPRM instrumentation trip channels, each trip channel consisting of two OPRM instrumentation modules. Each OPRM instrumentation module receives input from LPRMs. Each OPRM instrumentation module also receives input from the RPS Average Power Range Monitor (APRM) power and flow signals to automatically enable the trip function of the OPRM instrumentation module.

Each OPRM instrumentation module is continuously tested by a self-test function. On detection of any OPRM instrumentation module failure, either a "Trouble" or "INOP" alarm is activated. The OPRM instrumentation module provides an "INOP" alarm when the self-test feature indicates that the OPRM instrumentation module may not be capable of meeting its functional requirements. When one OPRM instrumentation module is inoperable, the remaining redundant OPRM Instrumentation module in the associated OPRM trip channel maintains the operability of the trip channel and thus there is no loss of trip function redundancy. The OPRM Instrumentation System provides compliance with GDC 10 and GDC 12.

The incorporation of the OPRM instrumentation into the TS will allow the deletion of the current thermal hydraulic instability administrative requirements and Power versus Flow TS Figure and associated actions. The OPRM instrumentation will provide the same level of assurance that the MCPR safety limit will not be violated for anticipated oscillations as that provided by the Power versus Flow TS Figure.

The OPRM Instrumentation System enabled region of the Power versus Flow figure was adjusted to maintain the same level of protection against the occurrence of a thermal-hydraulic instability by maintaining the pre-power uprate absolute power and flow coordinates. A 5% Power Uprate was approved for LaSalle County Station, Units 1 and 2, by Facility Operating License Amendments 140 and 125, respectively, in an NRC letter dated May 9, 2000.

The proposed changes do not affect the margin of safety as the OPRM Instrumentation will initiate an automatic reactor trip upon detection of an instability that could threaten the MCPR safety limit.

Thus, this proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: September 29, 2000.

Description of amendment request: The proposed amendments would make various changes to the Technical Specifications (TSs) to support a change in fuel vendors from Siemens Power Corporation to General Electric and a transition to the use of General Electric (GE) 14 fuel.

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

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

Proposed Changes to Current Technical Specifications

Evaluation of effect on the probability of an accident previously evaluated:

1. *Administrative Changes*. The revisions to Current Technical Specifications (CTS) Sections 2.1.B, ''Thermal Power, High Pressure and High Flow,'' and 3.6.A, ''Recirculation Loops,'' regarding the Minimum Critical Power Ratio (MCPR) Safety Limit, the changes to Section 3.11B, ''Transient Linear Heat Generation Rate,'' regarding the surveillance to monitor Transient linear heat Generation Rate (TLHGR) using either the ratio of the Maximum Fraction of Limiting Power Density (MFLPD) to the Fraction of Rated Thermal Power (FRTP) or the Fuel Design Limiting Ratio for Centerline (FDLRC) Melt, and the addition of the NRC approved RODEX2A methodology, are administrative changes and will not affect the probability of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology*. The changes to CTS Sections 3/4.3.C, ''Control Rod Operability,'' 3/4.3.D, ''Maximum Scram Insertion Times,'' 3/4.3.E, ''Average Scram Insertion Times,'' 3/4.3.F, ''Group Scram Insertion Times,'' 3/4.3.G, ''Control Rod Scram Accumulators,'' 3/4.3.H, ''Control Rod Coupling,'' and 3/4.3.I, ''Control Rod Position Indication System,'' revise the methodology for determining rod operability and control rod scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not affect the probability of an accident previously evaluated.

3. *Control Rod Scram Times*. The addition of required scram times for General Electric (GE) analyzed cores does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. *Administrative Changes*. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit are administrative changes and will not affect the consequences of an accident previously evaluated. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes. The changes to this section are analytical in nature and do not affect plant systems, structures, or components. The administrative changes to Section 3.11.B revise the description of fuel thermal limits that are monitored to ensure the TLHGR limit is not violated. TLHGR protects the fuel from 1% plastic strain and fuel centerline melt. Because these criteria have not changed, the consequences of an accident have not changed. The NRC approved burnup extension for RODEX2A has been demonstrated to meet all applicable design criteria. Therefore, the addition of the NRC approved RODEX2A methodology does not increase the consequences of an accident previously evaluated.

2. *Control Rod Operability and Scram Insertion Times Methodology*. The revisions to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I are made to ensure that appropriate scram times are reflected in the TS for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and local scram times (*i.e.,* a four control rod group). The proposed revisions, based on the Improved Technical Specification (ITS) methods, ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

In addition, numerous changes to the control rod operability and scram timing TS Sections were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.

Administrative changes. These consist of restructuring, interpretation, rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the consequences of an accident previously evaluated.

More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

Less restrictive changes. The less restrictive changes involve increasing the time to complete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies are not assumed in the analysis of the consequences of any accidents previously evaluated, and therefore, cannot increase the consequences of such accidents. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. The revised scram timing methods will result in operating thermal limits that will maintain the identical safety limits. Thus, the consequences of the evaluated accidents will not increase.

3. *Control Rod Scram Times*. Cycle-specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific events in the Updated Final Safety Analysis Report (UFSAR), GE has determined that there is negligible impact on results of events which are not analyzed on a cycle-specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient.

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

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

1. *Administrative Changes*. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, the revisions to CTS Section 3.11.B to revise the description of TLHGR, and the addition of the NRC approved RODEX2A methodology are administrative changes and will not create the possibility of a new or different kind of accident. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology*. The changes to CTS Sections 3/4.3.C, 3/4.3.D, 3/4.3.E, 3/4.3.F, 3/4.3.G, 3/4.3.H, and 3/4.3.I revise the control rod operability and scram time requirements for operation. These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.

3. *Control Rod Scram Times*. These changes do not physically alter plant systems, structures or components and therefore do not create the possibility of a new or different kind of accident.

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

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

1. *Administrative Changes*. The revisions to CTS Sections 2.1.B and 3.6.A, regarding the MCPR Safety Limit, and the changes to CTS Section 3.11.B regarding the surveillance to monitor TLHGR, and the addition of the NRC approved RODEX2A methodology are administrative changes and will not reduce the margin of safety. These changes do not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Operability and Scram Insertion Times Methodology*. The revisions to the CTS control rod operability and scram insertion times ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. CTS methods ensure this by limiting scram times for individual rods, the average scram time, and local scram times (*i.e.,* a four control rod group). ITS methods ensure this by limiting the scram times for individual rods, the number of slow rods, and the number of adjacent slow rods. Each of these methods ensure equivalent protection of the assumed reactivity insertion rate. Therefore, the changes do not involve a reduction in the margin of safety.

In addition, numerous changes to the control rod operability and scram timing TS Sections were made to reflect the ITS approach to these requirements. These revisions consist of administrative changes, more restrictive changes, and less restrictive changes. The discussion of each of these categories is provided below.

Administrative Changes. These consist of restructuring, interpretation, and complex rearranging of requirements, and other changes not substantially revising an existing requirement. Therefore, these changes do not affect the margin of safety.

More restrictive changes. These consist of changes resulting in added restrictions or eliminating flexibility. The more restrictive requirements continue to ensure that process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, these changes do not reduce the margin of safety.

Less restrictive changes. The less restrictive changes involve increasing the time to compete actions, increasing the time intervals between required surveillances, and deleting or revising the applicability of certain actions. The time to complete actions and the surveillance frequencies have been extended for several reasons, including experience showing low probability of failures, the benefit of allowing time to perform actions without undue haste, or due to compensating changes in other actions. The deleted or revised actions are not assumed in the safety analyses for any evaluated accidents. Thus, there is no significant reduction in the margin of safety.

3. *Control Rod Scram Times*. The addition of required scram times for GE analyzed cores based on GE analysis methodology does not involve a reduction in the margin of safety. For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure the MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

Therefore, these proposed changes to the CTS do not involve a significant reduction in the margin safety.

Proposed Changes to ITS

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

Evaluation of the effect on the probability of an accident previously evaluated.

1. *Administrative change*. The addition of the NRC approved RODEX2A methodology is an administrative change and will not affect the probability of an accident previously evaluated. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by these changes.

2. *Control Rod Scram Times*. The revision to ITS Table 3.1.4–1, ‘‘Control

Rod Scram Times,'' adds scram time requirements for GE analyzed cores. This change does not physically alter plant systems, structures or components and therefore does not affect the probability of an accident previously evaluated.

Evaluation of the effect on the consequences of an accident previously evaluated.

1. *Administrative Change.* The NRC approved burnup extension for RODEX2A has been demonstrated to meet all applicable design criteria. Therefore, the addition of the NRC approved RODEX2A methodology does not increase the consequences of an accident previously evaluated.

2. *Control Rod Scram Times.* The revisions to ITS Section 3.1.4, ''Control Rod Scram Insertion Times,'' are made to ensure the appropriate scram times are reflected in the Technical Specifications (TS) for GE methodology. The scram timing requirements ensure that the negative reactivity insertion rate assumed in the safety analyses is preserved. Cycle specific analyses that use the GE methodology scram times will meet all of the same safety limit acceptance criteria. Additionally, for the non-cycle specific events in the UFSAR, GE has determined that there is negligible impact on the results of events which are not analyzed on a cycle specific basis. Therefore, there is no change to the consequences of a previously evaluated accident or transient due to the TS changes.

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

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

1. *Administrative Change.* The addition of the NRC approved RODEX2A methodology is an administrative change and will not create the possibility of a new or different kind of accident. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by this change.

2. *Control Rod Scram Insertion Times.* The revisions to ITS Section 3.1.4 do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes to these sections revise the control rod scram time requirements for operation. This changes does not physically alter plant systems, structures, or components.

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

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

1. *Administrative Change.* The addition of the NRC approved RODEX2A methodology is an administrative change and will not reduce the margin of safety. This change does not affect plant systems, structures, or components. No plant mitigating systems or functions are affected by this change.

2. *Control Rod Scram Insertion Times.* For GE analyzed cores, cycle-specific analyses using the actual averaged scram times provide MCPR operating limits that will ensure that MCPR safety limit is not violated. Therefore, the fuel remains appropriately protected and no margins of safety are reduced.

Therefore, the proposed changes to the ITS 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 requested amendments involve no significant hazards consideration.

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

NRC Section Chief: Anthony J. Mendiola.

Detroit Edison Energy Company, Docket No. 50–341, Fermi 2, Monroe County, Michigan

Date of amendment request: November 21, 2000.

Description of amendment request: The proposed amendment would approve a proposed change to the licensing basis regarding the timing of the release of fission products following an accident. The proposed change is based upon one of the insights established in NUREG–1465, ''Accident Source Terms for Light Water Nuclear Power Plants,'' which recognizes that there is a delay in the release of fission products from the reactor fuel following a postulated design-basis loss-of-coolant accident (LOCA). The timing of fission product release from perforated fuel rods (*i.e.,* the gap activity release) is based on the boiling-water reactor (BWR)-specific value of the timing of the gap activity release phase of a LOCA as calculated in the BWR Owners Group (BWROG) Report, ''Prediction of the Onset of Fission Gas Release From Fuel in Generic BWR,'' NEDC–32963A, dated March 2000, as previously approved by the NRC staff. This BWROG report would be added (as Reference 4) to the list of references in Updated Final Safety Analysis Report (UFSAR) Section 15.6.7. The licensing basis change to UFSAR Section 15.6.5.5.1, ''Fission Product Release From Fuel,'' would add the following: ''For primary containment isolation purposes, the activity from the damaged core is assumed to be released into the containment at 121 seconds following the accident. This timing assumption recognizes conclusions derived from the source term studies described in NUREG–1465, Regulatory Guide 1.183 and Reference 4. * * * The results of this Table [15.6.5–2, which presents the airborne activity in the containment] conservatively assume activity released from the core enters the drywell at accident time zero.'' UFSAR Section 15.6.5.5.2, ''Fission Product Transport to the Environment,'' would be similarly supplemented to state, ''The results in this Table [15.6.5–3, which gives the fission product release to the environment due to containment leakage and leakage from engineered safety feature components outside containment] conservatively assume activity released from the core enters the drywell at accident time zero.'' UFSAR Section 15.6.5.5.3, ''Results,'' would be supplemented to state, ''Dose associated with coolant activity release in the first 121 seconds of the accident is not included in this Table [15.6.5–4, which presents the calculated exposures for the design basis analysis]. Its contribution to the accident dose is insignificant (on the order of 2 rem [to the] thyroid at the Exclusion Area Boundary).''

The effect of the NRC staff's approval of the proposed amendment is to allow the licensee, in accordance with 10 CFR 50.59, to increase the automatic closure times for selected primary containment isolation valves (PCIVs) (*i.e.,* those PCIVs credited for limiting post-accident doses to both control room personnel and to offsite individuals). Valves with closure times based on other requirements (*i.e.,* system performance requirements, equipment qualification, high-energy line break mitigation, or other regulatory requirements) would not be affected by the proposed change.

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

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

The proposed change takes credit for one of the alternative source term (AST) insights contained in NUREG–1465 which recognizes that fission product release from a fuel assembly is not instantaneous in a design basis accident. Implementation of this change into the licensing basis will be used to justify an increase in the maximum allowable closure times for primary containment isolation valves. A change in the timing of the gap release does not affect the precursors for any accident or transient previously evaluated as part of the Fermi 2 licensing basis. Therefore, there is no increase in the probability of any accident.

A plant specific radiological analysis has been performed to evaluate the effects of extending the maximum allowable valve closure times on accident dose consequences. This evaluation utilized the insights contained in NUREG–1465 * * * and NEDC–32963A * * * to justify no gap activity release during the initial 121 seconds of the accident. Therefore, during this period, the only releases are from reactor coolant activity. Assuming the maximum coolant iodine activity permitted in the Technical Specifications, the 2-hour Exclusion Area Boundary (EAB) dose associated with this release has been conservatively estimated to be less than 2 rem thyroid. This dose represents a small fraction of the LOCA dose evaluated in the UFSAR and is significantly lower than the 300 rem thyroid dose acceptance limit in 10 CFR Part 100.

UFSAR Figures 6.2–9 and 6.2–11 show the DBA [design-basis accident] LOCA primary containment pressure response. These figures indicate that drywell pressure peaks at around 5 seconds into the accident before gradually dropping off; therefore, PCIVs would not be required to close against increased containment pressure as a result of this change.

Utilizing all of the insights contained in NUREG–1465, would result in a reduction in the calculated dose. However, because this request is for a selective implementation of the AST scope, crediting only the timing of the gap activity release, the long term dose calculations based on TID–14844 in the UFSAR are not changed. Therefore, it is concluded that the proposed change does not significantly increase the consequences of a previously evaluated accident.

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

The primary containment isolation system is designed to prevent the unfiltered release of radioactive material to the environs following an accident. Therefore, the system is relied upon to mitigate the dose consequences of an accident. The proposed change recognizes the time delay before fission products are released into the containment as a result of fuel damage and allows for the adjustment of the maximum PCIV closure times accordingly. This change does not affect the function of the primary containment isolation system. The relaxation in valve closure times will be applied only to valves that do not have other system performance requirements on isolation time. Therefore, the proposed change does not create the potential for a new or different kind of accident from any accident previously evaluated.

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

The proposed change revises the Fermi 2 licensing basis for the offsite dose calculations during the initial 121 seconds of a LOCA scenario. For this period of time, only coolant activity release is postulated. No fission product release from perforated fuel rods is assumed. All other assumptions, bases and methodologies used in the long-term offsite dose calculations remain unchanged. The total dose shown in UFSAR Table 15.6.5–4 does not significantly increase due to the delay in the fission product release. The total amount of radioactivity remains the same and is bounded by the limits established in 10 CFR 100. The dose associated with coolant activity release in the initial 121 seconds of the accident has been determined to be insignificant. Therefore, the proposed change will not result in a significant reduction in the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: November 22, 2000.

Description of amendment request: The proposed amendment would change the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's Technical Specifications (TSs) to cover operation between 20, 32, and 48 Effective Full Power Years. Also changes to the Bases section consistent with the TS changes are proposed.

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

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

The licensee has proposed to adopt a change in the calculation methodology for the pressure-temperature limits based upon Code Cases N–640 and N–588. The code cases were developed using knowledge gained through years of industry experience. Pressure-temperature curves developed using the allowances of Code Cases N–640 and N–588 yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the previous assumptions used for the calculation of pressure-temperature limits are overly conservative. There are no physical changes to the plant being introduced by the proposed changes to the pressure-temperature 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. Therefore, providing the allowances of the subject code cases in developing the pressure-temperature limit curves do not involve a significant increase in the probability or consequences of an accident previously evaluate. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes represent a change in the methodology in how the pressure-temperature curves were generated. The proposed changes provide more operating margin in the pressure-temperature limit curves for in-service leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality. However, compliance with the proposed pressure-temperature curves will ensure conditions in which brittle fracture of primary coolant pressure boundary materials is possible will be avoided because such compliance with the proposed pressure-temperature curves provides sufficient protection against a non-ductile-type fracture of the reactor pressure vessel. Therefore, no new modes of operation are introduced nor will the changes create any failure mode not bounded by the previously evaluated accidents. Further, the proposed changes to the pressure-temperature 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.

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

The proposed changes reflect an update of the pressure-temperature curves. The revised curves are based on the latest U.S. Nuclear Regulatory Commission and American Society of Mechanical Engineers (ASME) guidance. The revised pressure-temperature limits have been developed using the K_{ic} fracture toughness curve shown in the ASME Boiler and Pressure Vessel (B&PV) Code Section XI, Appendix A, Figure A–2000–1, in lieu of the K_{Ia} fracture toughness curve shown in ASME B&PV Code Section XI, Appendix G, Figure G–2010–1, as the lower bound fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Appendix G process of determining pressure-temperature limit curves remain unchanged.

These revised pressure-temperature limits, although less restrictive than the current limits, are established in accordance with current regulations and the latest ASME Code information. The revised pressure-temperature curves provide more operating margin and, thus, more operational flexibility than the current pressure-temperature curves. However, industry experience since the inception of the pressure-temperature limits in 1974 confirms that some of the original methodologies used to develop pressure-temperature curves are overly conservative. Accordingly, ASME Code Cases N–640 and N–588 take advantage of the acquired knowledge by establishing more realistic methodologies for the development of pressure-temperature curves. Therefore, operational flexibility is gained and an acceptable margin of safety to reactor pressure vessel non-ductile type fracture is maintained. No plant safety limits, setpoints, or design parameters are adversely affected by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 30, 2000.

Description of amendment request: The proposed amendment would relocate the boration systems requirements from the Technical Specifications (TSs) to the Technical Requirements Manual (TRM).

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

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

The boration systems, BAMT [boric acid makeup tank], Boric Acid Makeup Pumps, and Charging Pumps, are part of the CVCS [chemical and volume and control system], which functions to maintain Reactor Coolant System inventory and chemistry. The boration system functions will continue to be maintained in accordance with their associated design requirements. During accident conditions when a boration source is required for accident mitigation, the RWT [refueling water tank] provides suction for the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) pumps. The CVCS boration systems are not credited in the mitigation of any accidents. Therefore, the dose consequences associated with accident analysis will be unchanged. The HPSI, LPSI pumps and RWT are required by Technical Specifications.

Based on an evaluation of the criterion listed in 10 CFR 50.36(c)(2)(ii), the relocation of the CVCS boration systems to the TRM is acceptable. No changes will be made to these systems that will affect their current operation.

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

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

The design and functions of the Boric Acid Makeup Tanks, Boric Acid Makeup Pumps, Charging Pumps and associated flow paths will continue to be maintained. These systems are not accident initiators. Because the proposed amendment will not change the design, configuration or method of operation of the plant, it will not create the possibility of a new or different kind of accident.

Safety Analysis Report (SAR) Chapter 15 provides the analysis of accidents that are considered credible. The Uncontrolled Control Element Assemblies (CEA) withdrawal from a subcritical or a critical condition, Boration Dilution Event, and Loss of Coolant Accident (LOCA) were evaluated in relationship to relocating these specifications to the TRM. Boric acid injection via the CVCS system was not credited in mitigating any of these accidents.

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

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

The movement of these TSs to the TRM does not reduce the existing TSs or surveillance requirements. The proposed change does not change the design function for any of these components. Additionally, none of the boration systems contained in these specifications are credited in any accident analysis. The systems are used to maintain RCS [reactor coolant system] chemistry and inventory and this function will be maintained.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 10, 2000.

Description of amendment request: Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station (GGNS) Operating License be amended to modify those Technical Specifications (TS) required to support GGNS Cycle 12 operation. The modifications would include a change to the Safety Limit Minimum Critical Power Ratio (SLMCPR) reported in TS 2.1.1.2, and the references for analytical methods used to determine reactor core operating limits listed in TS 5.6.5. Specifically, the proposed amendment reflects a decrease of the two recirculation loop SLMCPR limit from 1.09 to 1.08, with the single recirculation loop SLMCPR limit remaining unchanged at 1.10. The proposed changes are necessary in order to reflect the Nuclear Regulatory Commission (NRC) approved methods used in determining the GGNS Cycle 12 core operating limits, and reflect the safety limit changes for the Cycle 12 mixed core consisting of Siemens Power Corporation (SPC) ATRIUM–10 reload fuel and General Electric (GE) GE–11 reactor fuel.

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

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

The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification 2.1.1 as that limit which ''ensures that during normal operation and during Anticipated Operational Occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling.'' The MCPR safety limit satisfies the requirements of General Design Criterion 10 of Appendix A to 10 CFR (Part) 50 regarding acceptable fuel design limits. The MCPR safety limit is re-evaluated for each reload using NRC-approved methodologies. The analyses for GGNS Cycle 12 have concluded that a two-loop MCPR safety limit of 1.08, based on the application of Siemens Power Corporation's NRC-approved MCPR safety limit methodology, will ensure that this acceptance criterion is met. For single-loop operation, a MCPR safety limit of 1.10 (unchanged), also ensures that this acceptance criterion is met.

In addition to the MCPR safety limit, core operating limits are established to support the Technical Specification 3.2 requirements which ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any anticipated operational occurrences (AOO). The methods used to determine the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in TS section 5.6.5. A change to TS section 5.6.5 is requested to include the SPC methods in the list of NRC approved methods applicable to GGNS. These NRC approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and in the event of an AOO.

The requested Technical Specification changes do not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigating systems, and do not introduce any new accident initiation mechanisms.

Therefore, these changes to the Minimum Critical Power Ratio (MCPR) safety limit and to the list of methods used to determine the core operating limits do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The ATRIUM–10 fuel to be used in Cycle 12 is of a design compatible with the co-resident GE–11. Therefore, the introduction of ATRIUM–10 fuel into the Cycle 12 core will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. The proposed revised MCPR safety limits have accounted for the mixed fuel core and have been shown to be acceptable for Cycle 12 operation. Compliance with the criterion for incipient boiling transition continues to be ensured. The core operating limits will continue to be developed using NRC approved methods which also account for the mixed fuel core design. The proposed MCPR safety limits or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident.

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

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

The MCPR safety limits have been evaluated in accordance with Siemens Power Corporation's NRC-approved cycle-specific safety limit methodology to ensure that during normal operation and during Anticipated Operational Occurrences (AOO's) at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. On this basis, the implementation of this Siemens Power Corporation methodology does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

FirstEnergy Nuclear Operating Company, et al., Docket Nos. 50–334 and 50–412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: November 8, 2000.

Description of amendment request: The proposed amendment will delete Technical Specification (TS) 3/4.4.1.6, ''Reactor Coolant Pump-Startup,'' from the Beaver Valley Power Station (BVPS) TSs. This is accompanied by moving the secondary side water temperature to cold leg temperature difference Reactor Coolant Pump (RCP) start requirement to existing Reactor Coolant System (RCS) TSs and deleting the pressurizer level requirement from Unit 1 TS 3/4.4.1.6. Unit 2 TS 3/4.4.1.6 does not contain the pressurizer level requirement. The RCS TSs affected are TS 3/4.4.1.2, ''Reactor Coolant System—Hot Standby,'' (for Unit 2 only) and 3/4.4.1.3, ''Reactor Coolant System—Shutdown,'' (both units).

Changes to the affected Bases of the Technical Specifications will also be made.

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

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

The proposed changes will not significantly increase the probability of an accident previously evaluated in the BVPS Updated Final Safety Analysis Report (UFSAR) because accident initiation probabilities are independent of these changes. The proposed changes do not adversely affect any accident initiating events. The assumptions of the safety analysis are not changed by this license amendment request. The applicable concern associated is the possibility of overpressurizing the Reactor Coolant System (RCS) when a Reactor Coolant Pump (RCP) is started in a non-isolated loop. Adhering to a maximum secondary to primary side temperature difference (Technical Specifications 3/4.4.1.2, Reactor Coolant System—Hot Standby, Unit 2 only, and 3/4.4.1.3, Reactor Coolant System—Shutdown, both units), before an RCP is started and the operability of the OPPS (Technical Specification 3/4.4.9.3, Overpressure Protection Systems, for both units), which uses the PORVs as a pressure relief device, prevents this. The existing Technical Specifications specify when the OPPS is to be operable, the maximum secondary to primary side temperature difference permitted, and the operability requirements imposed on the PORVs.

The consequences associated with the starting of an RCP and potential overpressurization of the RCS also are not changed by the proposed license amendment. None of the accident prevention or mitigation controls or capabilities have been changed. Reactor Coolant Pump start restrictions are retained with the Technical Specifications, except for the pressurizer level requirement for BVPS Unit 1. This requirement has been shown to be unnecessary in preventing RCS overpressurization because the analysis assumes a water solid pressurizer when at least one PORV is operable. The safety analysis has shown that the temperature difference requirement is sufficient to preclude RCS overpressurization provided one PORV is available for pressure relief. As a result, the proposed changes will not affect any accident analysis consequences.

The Technical Specifications continue to specify the maximum secondary to primary side temperature difference, when the OPPS is to be enabled, and the operability requirements for the PORVs. These requirements are not altered by this license amendment request and will continue to assure that the OPPS analysis assumptions are met. It is sufficient to specify the temperature difference restriction for only Unit 2 Technical Specification 3/4.4.1.2 because the Unit 1 OPPS enabling temperature is not within the applicability of Technical Specification 3/4.4.1.2; *i.e.,* Mode 3, whereas the OPPS enabling temperature is for Unit 2. Therefore, assurance is provided that the 10 CFR 50 Appendix G limits are not exceeded and that this proposed change is acceptable.

The Bases and editorial changes, needed to meet format requirements and reflect the deletion of Technical Specification 3/4.4.1.6, have no effect on accident probabilities or consequences.

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

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

The proposed changes do not modify the manner in which any plant equipment is maintained. The equipment used to prevent RCS overpressurization is not altered by the proposed changes. Specification of the number of PORVs required to be operable when the OPPS is enabled, and at what temperature the OPPS is required, will continue to be retained in Technical Specification 3/4.4.9.3, Overpressure Protection Systems. The necessary RCP start restrictions assumed in the safety analysis are not affected by the proposed changes. It has been shown that deleting the pressurizer level requirement for Unit 1 is consistent with the OPPS analysis. To assure the 10 CFR 50 Appendix G limits are not violated, the necessary requirements for starting an RCP in a non-isolated loop are retained within the Technical Specifications. Therefore, the analysis of an overpressurization of the RCS due to a heat input transient caused by starting an idle RCP is not changed by this license amendment request.

The Bases and editorial changes, needed to meet format requirements and reflect the deletion of Technical Specification 3/4.4.1.6, will not affect the creation of accidents. The OPPS analysis has demonstrated that an RCP can be started with a water solid RCS, provided the secondary to primary side temperature difference requirement is met, and a single PORV is available for pressure relief, without violating 10 CFR 50 Appendix G limits.

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

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

The margin of safety associated with starting an RCP in a non-isolated loop is the ability of a single OPPS PORV to relieve the potential RCS pressure increase without violating 10 CFR 50 Appendix G limits. This is maintained by meeting the secondary side water temperature to cold leg temperature difference and PORV operability requirements imposed by the Technical Specifications. These Technical Specification requirements are not altered by the proposed changes. The only deletion being proposed is the elimination of the pressurizer level requirement for BVPS Unit 1. This requirement has been shown to be unnecessary in meeting 10 CFR 50 Appendix G limits because the OPPS analysis assumes a water solid pressurizer and at least one OPPS PORV is operable. Starting an RCP with both OPPS PORVs not operable is not consistent with the RCS venting actions required by Technical Specification 3/4.4.9.3. In order to comply with the venting required actions with neither PORV operable, the RCS must be depressurized or in the process of being depressurized. Depressurization of the RCS would preclude starting an RCP. In order to start an RCP, the RCS must be pressurized to ensure a minimum pressure differential exists across the No. 1 seal of the RCP. Therefore, the PORV related requirements of Unit 1 Technical Specification 3/4.4.1.6 are sufficiently addressed by Technical Specification 3/4.4.9.3. By eliminating PORV operability requirements from Unit 1 Technical Specification 3/4.4.1.6, the Technical Specifications become more consistent between the two units and with the Standard Technical Specifications. All other RCP start and OPPS requirements are retained within the Technical Specifications and associated Bases sections.

The Bases and editorial changes, needed to meet format requirements and reflect the deletion of Technical Specification 3/4.4.1.6, will not affect the margin of safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety regarding meeting 10 CFR 50 Appendix G limits.

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, Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Marsha Gamberoni.

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

Date of amendment request: November 9, 2000.

Description of amendment request: The proposed changes would revise the action statements of the Davis-Besse Nuclear Power Station Technical Specifications (DBNPS) (TS) Limiting Condition for Operation (LCO) 3.5.2 and 3.6.2.1. This proposal would extend the allowed outage time for one Low Pressure Injection (LPI) System/Decay Heat Cooler train of an Emergency Core Cooling System (ECCS) subsystem from 72 hours to 7 days (168 hours) for LCO 3.5.2. One Containment Spray System train may be impacted by the inoperability of the associated LPI train. Therefore, an extension of the allowed outage time for one train of the Containment Spray System from 72 hours to 7 days for LCO 3.6.2.1 is also being proposed, as well as new information to be added to TS Bases Section 3/4.5.2 and 3/4.5.3 to clarify the TS LCO 3.5.2 requirements. These proposed changes are based on the Babcock & Wilcox Owners Group (BWOG) Topical report BAW–2295A, Revisions 1 & 2, ''Justification for the Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray System,'' dated October 9, 1998.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because, as demonstrated in the Babcock & Wilcox Owners Group's Topical Report BAW–2295A, Revisions 1 and 2, *Justification for Extension of Allowed Outage Time for Low Pressure Injection and Reactor Building Spray Systems,* no accident initiators, conditions, or assumptions are affected by the proposed changes to extend the allowed outage time (AOT) from 72 hours to 7 days for one inoperable train of Low Pressure Injection (LPI) in Technical Specification (TS) 3/4.5.2 Emergency Core Cooling Systems—ECCS subsystems—$T_{avg} \geq 280°F$ or Containment Spray in TS 3/4.6.2.1, Containment Systems—Depressurization and Cooling Systems—Containment Spray System. The proposed change to TS Bases Section 3/4.5.2 and 3/4.5.3 are discussions of the present TS Limiting Condition for Operation (LCO) which do not affect the probability of an accident.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because an extension in the allowable outage time from 72 hours to 7 days for one inoperable train will not affect any previously evaluated accidents. The proposed changes to the TS Bases discuss the present TS LCO and do not affect the consequences of an accident. The proposed changes do not alter the source term, containment isolation, or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure mode or transient is introduced since the proposed changes do not involve a plant modification or allow operation of any plant systems, structures, or components in a manner not addressed in the DBNPS Design Basis Accident analyses.

3. Not involve a significant reduction in a margin of safety because extending the allowed outage time to 7 days for one inoperable train does not impact any assumptions or inputs in the DBNPS Updated Safety Analysis Report. The proposed changes have been evaluated and determined that the extended allowed outage time is consistent with safe operation considering the redundant systems of required features and the administrative controls in place for removing this equipment from service. The proposed TS Bases changes reflect the existing TS LCO and, therefore, do not reduce a margin of safety.

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 9, 2000.

Description of amendment request: The proposed change would relocate Technical Specification (TS) 3/4.4.9.2 to the Davis Besse Nuclear Power Station (DBNPS) Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). A corresponding change to the TS index is also proposed. Relocation of TS 3/4.4.9.2 to the USAR TRM will allow future proposed changes to the requirements to be evaluated in accordance with 10 CFR 50.59 and implemented if prior Nuclear Regulatory Commission (NRC) approval is not required. The proposed change is in accordance with the requirements of 10 CFR 50.36 and the relocation guidance provided in the NRC's ''Final Policy Statement on TS Improvements for Nuclear Reactors,'' dated July 22, 1993. The proposed change is also in accordance with the guidance provided by the improved ''Standard Technical

Specifications—Babcock & Wilcox Plants,'' NUREG–1430, Revision 1.

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

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

1a. Not involve a significant increase in the probability of an accident previously evaluated because no change is being made to any accident initiator. No previously analyzed accident scenario is changed, and initiating conditions and assumptions remain as previously analyzed.

The proposed change would relocate TS 3/4.4.9.2 ''Reactor Coolant System—Pressurizer,'' to the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM). TS 3/4.4.9.2 provides temperature limits for the Pressurizer based on its fatigue analysis design criteria. The proposed change to remove this TS is in accordance with 10 CFR 50.36 and the NRC's ''Final Policy Statement on TS Improvements for Nuclear Power Reactors,'' dated July 22, 1993. The proposed change is also consistent with the improved ''Standard Technical Specifications—Babcock and Wilcox Plants,'' NUREG–1430, Revision 1. A corresponding change to the TS Index page V that removes reference to the Pressurizer Pressure/Temperature Limits is an administrative change.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed change does not affect accident conditions or assumptions used in evaluating the radiological consequences of an accident. The proposed change does not alter the source term, containment isolation or allowable radiological releases.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new failure mode is introduced since the proposed relocation does not involve a modification or change in operation of any plant systems, structures, or components. No new, or different types of failures or accident initiators are introduced by the proposed change.

3. Not involve a significant reduction in a margin of safety because the proposed change is administrative in nature, consisting of the relocation of certain TS requirements into a licensee-controlled document, and has no bearing on the margin of safety which exists in the present TS or Updated Safety Analysis Report (USAR).

On the basis of the above, the DBNPS has determined that the License Amendment Request does not involve a significant hazards consideration. As this License Amendment Request concerns a proposed change to the Technical Specifications that must be reviewed by the Nuclear Regulatory Commission, this License Amendment Request does not constitute an unreviewed safety question.

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

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

NRC Section Chief: Anthony J. Mendiola.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of amendment request: October 30, 2000.

Description of amendment request: The proposed amendment would revise the St. Lucie Unit 1 Technical Specification (TS) 3.9.4, Containment Penetrations. TS 3.9.4 requires a personnel airlock (PAL) door to be closed during core alterations or movement of irradiated fuel within containment. The proposed change would allow both containment PAL doors to be open during core alterations and movement of irradiated fuel in containment provided: (a) that at least one personnel airlock door is capable of being closed; (b) the plant is in MODE 6 with at least 23 feet of water above the fuel; and (c) a designated individual is available outside the PAL to close the door. Operability of the containment PAL door includes the requirements that the door is capable of being closed and that any cables or hoses across the PAL door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

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

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

The proposed change to TS 3.9.4 would allow the containment personnel airlock (PAL) doors to be open during fuel movement or core alterations. Currently, a single PAL door is closed during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. The PAL is not an initiator of an accident. Whether the PAL doors are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the PAL doors to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 100. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the PAL doors after the containment is evacuated. The proposed change should significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment.

The changes being proposed do not affect assumptions contained in plant safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed change to Technical Specification 3.9.4, Containment Penetrations, affects a previously evaluated fuel handling accident. Both the current and the reanalyzed fuel handling accident analysis assume that all of the iodine and noble gases that become airborne within the containment escape and reach the site boundary and low population zone with no credit taken for filtration, the containment building barrier, or for decay or deposition taken. Since the proposed change does not involve the addition or modification of equipment, nor does it alter the design of plant systems and the revised analysis is consistent with the fuel handling accident analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined by 10 CFR Part 100 has not been reduced. The calculated dose is a well within of the limits given in 10 CFR Part 100 or NUREG–0800. The proposed changes do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 27, 2000.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated November 27, 2000.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: November 28, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 6.2.1, Minimum Shift Crew Composition with Two Separate Control Rooms and TS Section 6.3.1 (2), Unit Staff Qualifications for the Shift Technical Advisor (STA). The proposed amendments would permit, as an alternative to the current dedicated STA, an on-shift senior reactor operator (SRO) position to be combined with the required STA position. The proposed amendments would require an individual filling either the dedicated STA position or the combined SRO/STA position to meet the Technical Specifications educational requirements as described in **Federal Register** Notice 50 FR 43621, ''Commission Policy Statement on Engineering Expertise on Shift.'' These proposed changes are in accordance with the recommendations in the NRC *Policy Statement on Engineering Expertise on Shift,* published on October 28, 1985 and transmitted to all power reactor licensees and applicants by NRC Generic Letter 86–04, of the same title as the October 28, 1985 policy statement, dated February 13, 1986. As permitted by the policy statement, FPL proposes to exercise either of the STA options on a shift-by-shift basis.

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

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

Implementation of the proposed changes will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. Therefore, the proposed use of either the dual role SRO/STA position or the current dedicated STA position does not increase the probability of an accident previously evaluated. Implementation of the proposed changes will result in personnel with enhanced operational knowledge being assigned to perform the STA function of providing accident assessment expertise, and analyzing and responding to off normal occurrences when needed.

The NRC stated preference in the October 28, 1985, *Policy Statement on Engineering Expertise on Shift,* indicates that the NRC has concluded that the individual filling the dual role SRO/STA position may perform these functions better than a non-licensed individual filling the STA position, even when the SRO/STA is concurrently functioning as one of the required shift SROs. Therefore, the proposed TS changes do not increase the consequences of an accident previously evaluated.

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

The proposed amendments will not change the physical plant or the modes of plant operation defined in the facility license for either St. Lucie unit. Changes proposed for the administrative controls do not involve the addition or modification of equipment, nor do they alter the design or operation of plant systems. Therefore, operation of either facility in accordance with its proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendments revise certain administrative controls involving the on-site programmatic process for review and approval of plant procedures. Neither the scope, nor the requirement to establish, maintain, and implement procedures for activities that could affect nuclear safety are being changed.

The NRC stated preference in the October 28, 1985, *Policy Statement on Engineering Expertise on Shift,* indicates that the NRC has concluded that the individual filling the dual role SRO/STA position may perform these functions better than a non-licensed individual filling the STA position, even when the SRO/STA is concurrently functioning as one of the required shift SROs. Therefore, the proposed amendments should involve an enhancement in a margin on 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: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: October 30, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification 5.3.2 for Turkey Point Units 3 and 4 to extend the residual heat removal (RHR) pump allowed outage time (AOT) from 72 hours to 7 days to restore an inoperable

RHR pump to operable status. The proposed extension is based on the projected time required to replace a leaking or failed pump shaft seal, perform post-maintenance testing, and complete any additional corrective actions that may be needed to restore the pump to operable status. The extended RHR pump AOT will provide adequate time so that future seal repair activities are completed successfully in a safe manner.

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

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

The RHR system is part of the Emergency Core Cooling System. Inoperable RHR pumps are not accident initiators in any accident previously evaluated, and an extended AOT to restore operability of an inoperable RHR pump would not increase the probability of occurrence of accidents previously analyzed. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The RHR system is primarily designed to mitigate the consequences of the large Loss Of Coolant Accident (LOCA). In addition, the RHR system provides for primary system heat removal during unit shutdown conditions. The proposed changes do not affect any of the assumptions relative to accident initiators or accident response provided in the plant safety analyses. Accordingly, the consequences of accidents previously evaluated do not change.

A Probabilistic Safety Assessment (PSA) was performed to evaluate the impact of extending the allowed outage time on the RHR pump from 72 hours to 7 days. FPL concluded from the results of that assessment that the risk contribution of the AOT extension is very small, and that the net impact of the proposed amendment may be risk neutral.

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

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

The proposed change does not alter the design, physical configuration, or modes of operation of the plant. Plant configurations that are prohibited by Technical Specifications will not be created by the AOT extension. Therefore, the proposed activity does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The margin of safety associated with the Emergency Core Cooling System is established by acceptance criteria for system performance defined in 10 CFR 50.46. The proposed amendments will not change these acceptance criteria or the operability requirements for equipment that is used to achieve such performance as demonstrated in the plant safety analyses. Moreover, a Probabilistic Safety Assessment of the risk impact of extending the AOT for a single inoperable RHR pump has concluded that the risk contribution is very small, RHR system reliability can potentially be improved, and the net impact of the proposed change may be risk neutral. Therefore, the change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: December 6, 2000.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated December 6, 2000.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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: November 15, 2000.

Description of amendment requests: The proposed amendments would revise the Technical Specification (TS) 3.2.6, ''Allowable Power Level—APL,'' and TS 1.38, ''Allowable Power Level (APL),'' definitions of APL to remove a condition that limits APL to 100 percent of rated thermal power.

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

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

No new accident initiators or precursors are created by the proposed T/S changes. Reactor thermal power and power distribution within the reactor core are not initiators or precursors to any previously evaluated accident. There are no physical changes to the plant associated with the proposed T/S changes that would create any new accident initiators or precursors. Therefore, the proposed T/S changes do not increase the probability of occurrence of any accident previously evaluated.

Reactor thermal power up to the calculated value of APL ensures that the accident analysis results are not impacted by maintaining reactor core power distribution within prescribed limits. Since T/S 1.3 still contains a limitation on the maximum reactor thermal power allowed during normal operations, the normal overall operating limits for the reactor core are not changed. Accident analyses generally include a calorimetric error allowance of 2% or assume an initial power level of at least 102%. Using the additional limit on reactor thermal power based on APL ensures operation within the power distribution limits assumed in the accident analyses. Therefore, the proposed T/S changes do not affect operation of the reactor core and do not modify either the maximum acceptable reactor thermal power or the maximum allowed power distribution limits.

The proposed T/S changes do not change or alter the design criteria for the systems or components used to mitigate the consequences of any design basis accident. The reactor protection system (RPS), including reactor trips based upon overall reactor thermal power and power distribution within the reactor core, are not affected by the proposed T/S changes. The initial conditions of the accident analyses, including maximum reactor thermal power and worst-case power distribution within the reactor core, are not changed. As a result, the expected operation of the emergency core cooling systems (ECCS) are not affected by the proposed T/S changes. Radiological consequences of previously evaluated accidents are not increased, since overall reactor thermal power and power distribution limits are still maintained within the assumptions of the accident analyses, and operation of the RPS and ECCS is not affected. Therefore, the proposed changes do not increase the consequences of any accident and do not impact offsite dose considerations.

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

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

Reactor thermal power and power distribution within the reactor core cannot be an initiator or precursor to an accident. There are no physical changes to the plant associated with the proposed T/S changes that would create any new accident initiators or precursors. The proposed T/S changes do not degrade the reliability of any existing system, structure, or component. No new failure modes, malfunctions, or system interactions are created. The maximum steady state reactor core power level as defined by T/S 1.3 is not changed. The actual power distribution limits are not changed since the calculated value of APL is not changed. Therefore, the accident analyses assumptions and results are unchanged.

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

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

The proposed T/S changes do not change either the overall maximum reactor thermal power allowed, or the reactor core power distribution limits allowed. Maximum reactor thermal power remains limited by T/S 1.3. The calculated value of APL in T/S 3.2.6 is not changed, and remains as a control to ensure reactor core power distribution limits consistent with the accident analyses are satisfied. Therefore, safety margins related to power distribution limits are not affected. The proposed T/S changes do not affect any of the T/S safety limits or T/S limiting safety system settings, and RPS setpoints as defined by the T/S are not changed or affected.

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.

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

Date of amendment request: November 28, 2000.

Description of amendment request: The proposed amendment would establish technical specifications (TSs) for the emergency service water system. It would also revise TS 3.0 to include general requirements for system operability.

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

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

The EFT-ESW [emergency filtration train-emergency service water] System is not an accident initiator. The proposed amendment provides operability requirements and surveillance requirements to ensure the ESW System is available and operable when required for accident mitigation. The proposed operability requirements and allowed outage times are consistent with similar requirements for the systems supported by the EFT-ESW System. Dose to the public and the Control Room operators are not affected by the proposed change. The proposed general LCO [limiting condition for operation] provides direction with respect to actions to be taken when support systems are inoperable.

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, the proposed amendment will not significantly increase the probability or the consequences of an accident previously evaluated.

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

The proposed Technical Specification change does not introduce new equipment operating modes, nor does the proposed change alter existing system relationships. The proposed amendment does not introduce new failure modes. The proposed amendment does not alter the equipment required for accident mitigation and considers the effects on supported systems when a support system is inoperable. When support systems are inoperable, actions are specified to be taken consistent with safe plant operation.

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

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

The proposed amendment provides specifications for the EFT–ESW System which are consistent with current Technical Specification requirements for other equipment. The proposed changes ensure that the EFT–ESW and other support systems will be available when required and provides adequate alternative actions when the support systems are not available. The allowed outage times for the EFT-ESW Pumps are consistent with that allowed for other equipment that would have similar importance to accident mitigation. The proposed general LCO does not result in a significant reduction in the margin of safety since it imposes requirements already in technical specifications for support systems included in technical specifications. In cases where support systems [are] not included in technical specifications, the proposed general LCO does not apply and actions determined to be required by the technical specifications will be taken for the supported systems.

Therefore, the proposed amendment 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 request involves no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: September 5, 2000.

Description of amendment request: The proposed amendment would revise the Fort Calhoun Station, Unit No. 1 (FCS) Technical Specifications (TS) to change the definition section, TS

Sections 2.10, 3.10, and 5.9, and the Bases of TS 1.1 and 1.3, to allow the use of nuclear fuel fabricated by Siemens Power Corporation at FCS. The definition of unrodded planar radial peaking factor (F_{xy}) and TS 2.10.4(3) are being deleted and TS 3.10 is being revised to reflect the deletion of this peaking factor. TS 5.9.5 is being revised to incorporate NRC-approved methodologies necessary to determine core operating limits with nuclear fuel from Siemens Power Corporation. The Bases to TS 1.1 and 1.3 are being revised to delete the discussion of the CE–1 correlation that is currently used to calculate minimum departure from nucleate boiling ratio and the value calculated by this method.

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

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

The proposed amendment is to incorporate Siemens Power Corporation topical reports for conducting reload analyses that have been previously reviewed and approved by the NRC. The applicable FCS Technical Specifications (TS) supported by these topical reports are being revised. These changes are necessary to support using nuclear fuel supplied by Siemens Power Corporation.

It is proposed to revise the Bases of TS 1.1 and 1.3 to reflect changes in methodologies for calculating the minimum Departure from Nucleate Boiling Ration (DNBR). The proposed methodology for determining the minimum DNBR for fuel supplied by Siemens Power Corporation is the NRC-approved EMF–92–153(P)(A) and Supplement 1, HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel. As stated in the Basis of TS 1.1, Fort Calhoun Station currently uses the NRC-approved CE–1 correlation with a minimum DNBR value of 1.18, which provides a 95% probability at a 95% confidence level that DNB will not occur for any operating condition. For Siemens fuel, using the HTP correlation with a minimum DNBR of 1.14, as proposed, will continue to provide a 95% probability at a 95% confidence level that DNB will not occur during any operating condition. The CE–1 correlation is more restrictive than the HTP correlation that will be used to predict the minimum DNBR limits for the Siemens fuel. For a given set of reactor coolant conditions, the CE–1 correlation provides a lower critical heat flux than the HTP correlation. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

It is proposed that the total planar radial peaking factor, F_{xy}^T, be eliminated from the Technical Specifications. The current need for this parameter is to protect assumptions about the maximum amount of planar peaking in the core. The limitation on the total planar radial peaking factor, F_{xy}^T, is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density—High, Limiting Conditions for Operation, and Limiting Safety Systems Settings set-points remain valid during operation. In a two-dimensional set-point analysis, as currently conducted, F_{xy}^T is combined with the maximum axial power profile (F_z) to produce the maximum allowable peaking factor (F_q) or equivalent Linear Heat Rate. This ensures conservative operation relative to assumptions on linear heat rate used as input to the loss of coolant accident and other transient analyses. In a three-dimensional analysis, as proposed with the use of Siemens methodology, these peaks are calculated directly during a series of pre-determined maneuvers (axial shape oscillation, power maneuver, or other transient).

Direct calculation of these peaks negates the need to make inferences about the amount of planar radial peaking that occurs in any particular plane within the core. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

It is proposed to add NRC-approved methodologies from Siemens Power Corporation to TS that are necessary to evaluate core parameters. The proposed additions of NRC-approved topical reports to the TS do not modify the manner in which the topical reports may be implemented. The core operating limits will continue to be determined using NRC-approved analytical methods. The plant will continue to operate within the limits specified by the Core Operating Limits Report and will take corrective actions as required by the current Technical Specifications should these limits be exceeded. Therefore, these changes will not significantly 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.

No new or different modes of operation are proposed as a result of these changes. The proposed revisions do not change any equipment required to mitigate the consequences of an accident. The proposed additions of NRC-approved topical reports to the TS do not modify the manner in which the topical reports may be implemented. The plant will continue to operate within the limits specified by the Core Operating Limits Report and will take corrective actions as required should these limits be exceeded. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

As required by TS 5.9.5, the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The proposed changes incorporate methodologies applicable for use with fuel supplied by Siemens Power Corporation that have been approved by the NRC as documented by Safety Evaluation Reports. Technical Specification 5.9.5 also requires that the core operating limits shall be determined so that all applicable limits of the safety analysis are met. These requirements will continue to be met. Therefore, OPPD concludes that 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: Perry D. Robinson, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: October 18, 2000.

Description of amendment request: The proposed amendment would revise the Fort Calhoun Station Unit 1 (FCS) Technical Specifications (TSs) to (1) extend the validity of the existing TS Figure 2–1A (RCS [reactor coolant system] Pressure-Temperature Limits for Heatup) and Figure 2–1B (RCS Pressure-Temperature Limits for Cooldown) from

20.0 effective full power years (EFPY) to 24.25 EFPY, (2) delete Figure 2–3 (Predicted Radiation Induced NDTT [nil ductility transition temperature] Shift), and (3) provide replacement guidance in TSs 2.1.2(6)(a) and (b) for use of the most current fluence analysis and Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials,'' for projecting reference temperature nil ductility (RT_{NDT}) at 24.25 EFPY. The proposed amendment would also revise the associated Bases section of TS 2.1.2.

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

The NRC previously approved Technical Specification Amendment No. 161 in March 1994 for the use of RCS Pressure-Temperature (P–T) Limits good to 20.0 EFPY. The proposed changes in this submittal reflect the validity of these same curves from 20.0 EFPY to 24.25 EFPY based on the implementation of extreme low radial leakage fuel management in 1992 (Cycle 14). Significant reductions in the fast neutron flux to the limiting 3–410 axial weld in the Fort Calhoun Station reactor pressure vessel were obtained, thus significantly increasing the time to when the fast neutron fluence input to the derivation of the previously approved P–T curves will be reached. Since no inputs (including assumed material properties of the limiting weld) to the existing analysis are being changed, extension of the validity of the curves from 20.0 EFPY to 24.25 EFPY is justified. In addition, deletion of Figure 2–3 and references to it are proposed. This proposed change removes an outdated figure which is non-operational in nature. The application of the current Regulatory Guide 1.99, Revision 2 is more appropriate for these purposes. Administrative changes to the Basis section of TS 2.1.2 are proposed to reflect the extension to 24.25 EFPY.

No accidents previously analyzed are affected by these changes, and it can be concluded that there is no 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 proposed changes do not physically alter the configuration of the plant and no new or different mode of operation is proposed. Extending the validity of the P–T curves more accurately projects reactor vessel embrittlement by accounting for improvements in FCS fuel management which have significantly reduced the fast neutron fluence to the limiting 3–410 axial weld, incorporates improved operating cycle efficiency, and applies the WCAP–15443, Revision 0 fluence analysis. The revised fluence analysis uses the ENDF/B–VI Nuclear Cross Section Library. Deletion of Figure 2–3 represents a change which does not affect plant operations. Figure 2–3 is administrative in nature, and proposed revisions to Specifications 2.1.2(6)(a) and (b) provide guidance consistent with the current Regulatory Guide for P–T curves updates. Update of the Technical Specification 2.1.2 Basis section represents an administrative change that does not affect plant operation.

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

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

The proposed changes to extend the validity of Technical Specification Figures 2–1A and 2–1B to 24.25 EFPY are consistent with the extreme low radial leakage fuel management implemented in 1992 (Cycle 14) and performance/application of the updated fluence analysis described above. With no changes to the inputs of the existing P-T limits analysis, there is no reduction in the margin of safety. Figure 2–3 is not used to provide limits on plant operation, and deletion of this figure, which uses a pre-Regulatory Guide 1.99, Revision 2 embrittlement correlation, is considered an improvement in the consistency of the requirements outlined in the Technical Specifications. This Figure is not used in plant operation and provides only a general indication of the RT_{NDT} shift. The TS 2.1.2 Basis section changes are administrative in nature and do not affect the margin of safety. The changes serve to maintain consistency with the NRC approval of Amendment No. 161.

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

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

Attorney for licensee: Perry D. Robinson, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: October 27, 2000.

Description of amendment request: The proposed amendment would revise Section 3.7 of the Fort Calhoun Station Unit 1 Technical Specifications to eliminate item 3.7(4) ''13.8 Kv Transmission Line'' which states: ''The 13.8 Kv transmission line will be energized and loaded to minimum shutdown requirements at each refueling outage following installation.''

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.

Eliminating the 13.8 kV testing requirement would have no impact upon the probability of an accident previously evaluated. The circuit breaker connecting the 13.8 kV power supply to the station electrical busses is normally open, so this power supply could not play a role in the initiation of any accident.

Eliminating the 13.8 kV testing requirement would have no impact upon the consequences of an accident previously evaluated. Existing accident analyses take no credit for the 13.8 kV power supply.

The 13.8 kV power supply is not credited for mitigation of licensing basis transients or postulated events added to the USAR [Updated Safety Analysis Report] by NRC requirements, such as Station Blackout (SB0).

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

The 13.8 kV power supply is only capable of supplying a limited number of components in the unlikely event that 161 kV, 345 kV, and the diesel-generators are unavailable. Eliminating the 13.8 kV testing requirement would 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.

Testing of the 13.8 kV power supply, as described in Technical Specification 3.7(4), is unrelated to any margin of safety. Therefore, deletion of the testing requirement will not reduce any 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: Perry D. Robinson, Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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: November 30, 2000.

Description of amendment requests: The proposed license amendments would change Technical Specification Section 3.5.1, ''Accumulators,'' by revising the limits for accumulator borated water volume (Surveillance Requirement (SR) 3.5.1.2) and nitrogen cover pressure (SR 3.5.1.3) to reflect analysis limits. These TS currently reflect nominal limits. These amendments are revising TS values consistent with other similar TS parameters which will aid in future clarity.

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 accumulators only function following an accident. They cannot initiate an accident. The proposed changes have no impact to plant operation and are administrative in nature. Changing the technical specification (TS) limits for accumulator volume and pressure from nominal to analysis values will provide greater consistency within the TS. Changing the volume limits to cubic feet verse[u]s percent level will eliminate any potential for future revision of these limits because of instrument tap relocation.

Plant parameters will continue to be administratively controlled within the allowed analysis parameters. The proposed limits for tank volume and nitrogen cover pressure are consistent with analysis values documented in the Final Safety Analysis Report and assume that the accident consequences remain unchanged.

There are no hardware changes or changes in the method by which any safety-related plant system performs its safety function.

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 accumulators only function following an accident. They cannot initiate an accident. The proposed changes are administrative in nature.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The changes are administrative in nature so there are no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are [sic] introduced.

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

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

The proposed changes are administrative in nature.

The proposed changes do not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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 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: November 30, 2000.

Description of amendment requests: The proposed license amendments would change the administrative controls sections of Technical Specification (TS) 5.5.14b and 5.5.14b.2 to incorporate the changes made to 10 CFR Part 50, Section 50.59. The proposed amendments would replace the word ''involve'' with ''require'' in TS 5.5.14b and revise TS 5.5.14b.2 to delete the reference to ''unreviewed safety question'' and restate the requirement as ''a change to the updated Final Safety Analysis Report or Bases that requires NRC approval pursuant to 10 CFR 50.59.''

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

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

The proposed change replaces the word ''involve'' with ''require'' and deletes reference to the term ''unreviewed safety question'' consistent with 10 CFR [Part 50, Section] 50.59. Deletion of the term ''unreviewed safety question'' was approved by the NRC with the revision to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the Technical Specification (TS) Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected.

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 proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. These changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements in the TS.

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

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

The proposed changes will not reduce the margin of safety because they have no effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval. The proposed changes to TS 5.5.14 are considered administrative in nature based on the revision to 10 CFR 50.59.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment 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 Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: October 25, 2000.

Description of amendment request: The amendment revises the Action Statements associated with Technical Specifications (TSs) Table 3.3.7.5–1 (''Accident Monitoring Instrumentation'') concerning the Drywell Hydrogen/Oxygen (H_2/O_2) Concentration Analyzers, and the associated TS Bases. PECO Energy proposes to add new Action Statements 82a and 82b concerning channel operability, which will replace the current requirements of Action Statements 80a and 80b, respectively, for the Drywell Hydrogen/Oxygen Concentration Analyzers.

Under the existing TS Action Statements for Table 3.3.7.5–1 (''Accident Monitoring Instrumentation''), with the number of operable accident monitoring instrumentation channels less than the ''required'' number of channels (quantity 2), restore the inoperable channels within 7 days or be in at least hot shutdown within the following 12 hours (Action Statement 80a). Additionally, with the number of operable accident monitoring instrumentation channels less than the ''minimum'' number of channels (quantity 1), restore the inoperable channel(s) within 48 hours or be in at least hot shutdown within the following 12 hours (Action Statement 80b).

Proposed Action Statement 82a for Table 3.3.7.5–1 will extend the duration from 7 to 30 days for less than the ''required'' number operable of channels. Additionally, the proposed Action Statement 82a will require that if the operable channel(s) cannot be restored within the 30 days, then a Special Report shall be provided to the NRC within the following 14 days.

Proposed Action 82b for Table 3.3.7.5–1 will extend the duration from 48 hours to 72 hours for less than the ''minimum'' number of operable channels. If the inoperable channel(s) cannot be restored with the 72 hours, then be in hot shutdown with the next 12 hours.

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

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

The proposed TS changes modify the Action Statements associated with the duration that the Drywell Hydrogen/Oxygen Concentration Analyzers can be inoperable. The Drywell Hydrogen/Oxygen Concentration Analyzers are not accident initiating equipment and are monitoring devices required to be available for monitoring hydrogen and oxygen following a LOCA. These analyzers do not perform any automatic or control functions. Therefore, the proposed changes will not increase the probability of an accident previously evaluated.

In the event of a failure of the Drywell Hydrogen/Oxygen Concentration Analyzers following a LOCA, concentrations of hydrogen and oxygen can be measured by utilizing grab samples with the post-accident sampling system. A single failure of either analyzer package would render that affected package inoperable with the redundant package fully capable of performing the required function at full capacity. Following a postulated LOCA, the hydrogen recombiners will be utilized to ensure that the oxygen concentration in the primary containment is maintained below the lower flammability limit as required by plan emergency procedures.

The extended completion times are based on the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), the low probability of an event requiring post-accident instrumentation during this interval, and the availability of alternate means to obtain the required information. Therefore, the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed technical specification changes modify the Action Statements associated with the duration that the Drywell Hydrogen/Oxygen Concentration Analyzers can be inoperable. They do not change the design or configuration of the plant. The Drywell Hydrogen/Oxygen Concentration Analyzers are not accident initiating equipment, and are monitoring devices required to be available for monitoring hydrogen and oxygen following a LOCA. The proposed changes do not create a system-level failure mode different than those that already exist. In addition, there are no operation or failure modes of the Drywell Hydrogen/Oxygen Concentration Analyzers that are accident initiators. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes in Action Statements do not affect any safety limits or analytical limits. There are also no changes to accident of transient core thermal hydraulic conditions, minimum combustible concentration limits, or fuel or reactor coolant boundary design limits, as a result of these proposed changes. The proposed Technical Specification changes modify the Action Statements associated with the duration that the Drywell Hydrogen/Oxygen Concentration Analyzers can be inoperable. Therefore, the proposed changes do not involve a significant reduction in the 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, Senior V.P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: November 29, 2000.

Description of amendment request: The proposed amendment would revise the Technical Specifications to reflect the enabling of the Oscillation Power Range Monitor (OPRM) instrumentation reactor protection system (RPS) trip function. The OPRM is designed to detect the onset of reactor core power oscillations resulting from thermal-hydraulic instability and suppresses them by initiating a reactor scram via the RPS trip logic.

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

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of Interim Corrective Actions (ICAs) and certain limiting conditions of operation of Technical Specifications (TS) 3.4.1. The OPRM system can automatically detect and suppress conditions necessary for thermal-hydraulic (T–H) instability. A T–H instability event has the potential to challenge the Minimum Critical Power Ratio (MCPR) safety limit. The restrictions of the ICAs and TS 3.4.1 were imposed to ensure adequate capability to detect and suppress conditions consistent with the onset of T–H oscillations that may develop into a T–H instability event. With the installation of the OPRM System, these restrictions are no longer required.

The probability of a T–H instability event is most significantly impacted by power to flow conditions such that only during operation inside specific regions of the power to flow map, in combination with power shape and inlet enthalpy conditions, can the occurrence of an instability event be postulated to occur. Operation in these regions may increase the probability that operation with conditions necessary for a T–H instability can occur.

However, when the OPRM is operable with operating limits as specified in the COLR [Core Operating Limits Report], the OPRM can automatically detect the imminent onset of local power oscillations and generate a trip signal. Actuation of an RPS trip will suppress conditions necessary for T–H instability and decrease the probability of a T–H instability event. In the event the trip capability of the OPRM is not maintained, the proposed change includes actions which limit the period of time before the effected OPRM channel (or RPS system) must be placed in the trip condition. If these actions would result in a trip function, an alternate method to detect and suppress thermal hydraulic oscillations is required. In either case the duration of this period of time is limited such that the increase in the probability of a T–H instability event is not significant. Therefore the proposed change does not result in a significant increase in the probability of an accident previously evaluated.

An unmitigated T–H instability event is postulated to cause a violation of the MCPR safety limit. The proposed change ensures mitigation of T–H instability events prior to challenging the MCPR safety limit if initiated from anticipated conditions by detection of the onset of oscillations and actuation of an RPS trip signal. The OPRM also provides the capability of an RPS trip being generated for T–H instability events initiated from unanticipated but postulated conditions. These mitigating capabilities of the OPRM system would become available as a result of the proposed change and have the potential to reduce the consequences of anticipated and postulated T–H instability events. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1. The OPRM system uses input signals shared with APRM and rod block functions to monitor core conditions and generate an RPS trip when required. Quality requirements for software design, testing, implementation and module self-testing of the OPRM system provide assurance that no new equipment malfunctions due to software errors are created. The design of the OPRM system also ensures that neither operation nor malfunction of the OPRM system will adversely impact the operation of other systems and no accident or equipment malfunction of these other systems could cause the OPRM system to malfunction or cause a different kind of accident. Therefore, operation with the OPRM system does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation in regions currently restricted by the requirements of ICAs and TS 3.4.1 is within the nominal operating domain and ranges of plant systems and components for which postulated equipment and accidents have been evaluated. Therefore operation within these regions does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change which specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in certain regions of the power to flow map does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in regions of the power to flow map currently restricted by the requirements of ICAs and TS 3.4.1.

The OPRM system monitors small groups of LPRM signals for indication of local variations of core power consistent with T–H oscillations and generates an RPS trip when conditions consistent with the onset of oscillations are detected. An unmitigated T–H instability event has the potential to result in a challenge to the MCPR safety limit. The OPRM system provides the capability to automatically detect and suppress conditions which might result in a T–H instability event and thereby maintains the margin of safety by providing automatic protection for the MCPR safety limit while significantly reducing the burden on the control room operators. In the event the trip capability of the OPRM is not maintained, the proposed change includes actions which limit the period of time before the effected OPRM channel (or RPS system) must be placed in the trip condition. If these actions would result in a trip function, an alternate method to detect and suppress thermal hydraulic oscillations is required. Since, in either case, the duration of this period of time is limited so that the increase in the probability of a T–H instability event is not significant. Operation with the OPRM system does not involve a significant reduction in a margin of safety.

Operation in regions currently restricted by the requirements of ICAs and TS 3.4.1 is within the nominal operating domain assumed for identifying the range of initial conditions considered in the analysis of anticipated operational occurrences and postulated accidents. Therefore, operation in these regions does not involve a significant reduction in the margin of safety.

The proposed change, which specifies limiting conditions for operations, required actions and surveillance requirements of the OPRM system and allows operation in certain regions of the power to flow map, does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of application request: November 21, 2000 (ULNRC–04346)

Description of amendment request: The proposed amendment request would change Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' of the Technical Specifications. The change would add Surveillance Requirement (SR) 3.3.2.10 to the SRs for the following two engineered safety feature actuation system (ESFAS) instrumentation in the table: item f, loss of offsite power, and item h, auxiliary feedwater pump suction transfer on suction pressure—low. The licensee also identified that there would be changes to the Final Safety Analysis Report (FSAR).

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.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change imposes more stringent surveillance testing requirements to ensure safety-related structures, systems, and components are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance testing requirements. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change.

This change does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

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

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FdeltaH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan [(NUREG–0800)] will continue to be met.

The imposition of more stringent surveillance requirements [in the change] increase the margin of safety by ensuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

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 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: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of application request: November 22, 2000.

Description of amendment request: The proposed change to Callaway Technical Specification (TS) 5.5.14, which ensures that a program exists for processing changes to the TS Bases, would replace the word ''involve'' with ''require'' and deletes the phrase ''unreviewed safety question'' as defined in 10 CFR Part 50, Section 50.59.

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

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

The proposed changes replace the word ''involve'' with ''require'' and deletes the phrase ''unreviewed safety question'' as defined in 10 CFR 50.59. The above changes are consistent with the revision to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the Technical Specification Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not affected.

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 proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. These changes are considered administrative changes and do not modify, add, delete, or relocate any technical requirements in the TS.

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

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

The proposed changes will not reduce the margin of safety because they have no effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in paragraph (c)(2) of 10 CFR 50.59 will still require NRC approval. The proposed changes to TS 5.5.14 are considered administrative in nature based on the revisions to 10 CFR 50.59.

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

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

Attorney for licensee: John O'Neill, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: December 7, 2000 (ET 00–0041).

Description of amendment request: The proposed amendment request would change Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation,'' of the Technical Specifications (TSs). The change would add Surveillance Requirement (SR) 3.3.2.10 to the SRs for the following two engineered safety feature actuation system (ESFAS) instrumentation in the table: item 6.f, loss of offsite power, and item 6.h, auxiliary feedwater pump suction transfer on suction pressure—low. The licensee also identified that there would be changes to the Updated Safety Analysis Report (USAR) and changes to the Bases for the TSs.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change imposes more stringent surveillance testing requirements to ensure safety related structures, systems, and components are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the USAR.

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

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

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance testing requirements. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety related system as a result of this change.

This change does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

Therefore, the proposed changes do 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.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (FdeltaH), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan

[(NUREG–0800)] will continue to be met.

The imposition of more stringent surveillance testing requirements [in the change] increases the margin of safety by ensuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

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

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

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

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

Date of application for amendment: March 19, 1999, and supplemented by letters dated April 17, May 5, June 16, July 26, and November 21, 2000.

Brief description of amendment: The amendment changes Technical Specification (TS) 1.40, ''Spent Fuel Pool Storage Pattern''; 1.41, ''3–OUT–OF–4 AND 4–OUT–OF–4''; 3/4.9.1.2, ''Boron Concentration''; 3/4.9.7, ''Crane Travel-Spent Fuel Storage Areas''; 3/4.9.13, ''Spent Fuel Pool—Reactivity''; 3.9.14, ''Spent Fuel Pool—Storage Pattern''; 5.6.1.1, ''Design Features—Criticality''; and 5.6.3, ''Design Features—Capacity.'' In addition, the amendment revises INDEX pages xii and xv for new figures and page numbers and replaces Figures 3.9–1 and 3.9–2 with four new figures and make changes to the TS Bases consistent with changes to their respective TS sections.

Date of issuance: November 28, 2000.

Amendment No.: 189.

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

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

Date of individual notice in **Federal Register:** December 4, 2000 (65 FR 75736).

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).

Calvert Cliffs Nuclear Power Plant, Inc., 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 22, 1999, as supplemented November 24, 1999 and September 12, 2000.

Brief description of amendments: The amendments revise Technical Specification 5.5.11, ''Ventilation Filter Testing Program'' for laboratory testing of charcoal in engineered safety feature ventilation systems to reference American Society for Testing and Materials D3803–1989 ''Standard Test Method for Nuclear-Grade Activated Carbon.''

Date of issuance: December 7, 2000.

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

Amendment Nos.: 238 and 212.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 29, 1999 (64 FR 73085)

The November 24, 1999, and September 12, 2000, submittals 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 December 7, 2000.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., 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: September 15, 2000.

Brief description of amendments: The amendments implement Technical Specification Task Force (TSTF)–134, Revision 1. TSTF–134 revises Technical Specification Surveillance Requirements (SR) 3.1.7.2 which verifies control element assembly (CEA) trip function from 50 percent withdrawn position, by adding a note allowing SR 3.1.7.2 not be performed if TS SR 3.1.4.6 (CEA drop time test) has been met. TSTF–134, Revision 1, was approved by the Nuclear Regulatory Commission on April 21, 1998.

Date of issuance: December 11, 2000.

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

Amendment Nos.: 239 and 213.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 18, 2000 (65 FR 62384).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 5, 1999, as supplemented on December 22, 1999, and September 18, 2000.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.5, ''Instrumentation Systems,'' for the reactor protection system and engineered safety features actuation system instrumentation. Specifically, the amendment: (1) Revises the allowed outage times for the instrumentation, (2) allows on-line testing and maintenance of instrumentation, and (3) revises the associated Bases section. The amendment also includes several editorial changes to TS Tables 3.5–2 and 3.5–3.

Date of issuance: November 30, 2000.

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

Amendment No.: 212.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 4, 2000 (65 FR 59221).

The December 22, 1999, and September 18, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 30, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 22, 2000, as supplemented on October 3 and 15, 2000.

Brief description of amendment: The amendment revises: (1) Technical Specification (TS) 3.10.4, ''Rod Insertion Limits,'' to allow on-line calibration of the rod position indicator (RPI) channels during operating cycle 15, and (2) TS 3.10.6, ''Inoperable Rod Position Indicator Channels,'' to allow extended RPI deviation limits during cycle 15.

Date of issuance: December 12, 2000.

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

Amendment No.: 213.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 20, 2000 (65 FR 56948).

The October 3 and 15, 2000, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 12, 2000.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: February 21, 2000.

Brief description of amendment: This amendment deleted references to stainless steel as the material for reactor coolant system and reactor coolant pressure boundary component fasteners from Table 1.8–1 and 1.8–2 of the Beaver Valley Power Station, Unit No. 1, Updated Final Safety Analysis Report (UFSAR).

Date of issuance: December 4, 2000.

Effective date: As of date of issuance.

Amendment No.: 235.

Facility Operating License No. DPR–66: Amendment authorized changes to the UFSAR.

Date of initial notice in **Federal Register:** June 14, 2000 (65 FR 37426).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 4, 2000.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket No. 50–335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: July 19, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TS) surveillance requirements of the safety-related ventilation system charcoal consistent with the actions requested in Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999. Systems impacted include the control room emergency ventilation system, the shield building ventilation system, the emergency core cooling system area ventilation system, and the fuel pool ventilation system—fuel storage.

Date of Issuance: December 7, 2000.

Effective Date: December 7, 2000.

Amendment No.: 167.

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

Date of initial notice in **Federal Register:** August 9, 2000 (65 FR 48749).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 21, 2000.

Brief description of amendments: These amendments relocate Technical Specification (TS) Surveillance Requirement 4.8.1.1.2.e.1, regarding the emergency diesel generator (EDG) inspection program, to a licensee controlled maintenance program that will be incorporated by reference into the next revision of the Updated Final Safety Analysis Report for each St. Lucie unit. Upon relocation to the licensee controlled maintenance program, the effectiveness of the maintenance on the EDGs and support systems will be monitored pursuant to the Maintenance Rule 10 CFR 50.65.

Date of Issuance: December 7, 2000.

Effective Date: December 7, 2000.

Amendment Nos.: 168 and 111.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the TS.

Date of initial notice in **Federal Register:** August 9, 2000 (65 FR 48750).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 30, 1999, as supplemented June 28, 2000, and November 3, 2000.

Brief description of amendment: The amendment revises Technical Specifications Sections 3.7.2, ''Control Room Envelope Filtration (CREF) System,'' and 5.5.7, ''Ventilation Filter Testing Program (VFTP)'' for laboratory testing of charcoal filters to reference American Society for Testing and Materials standard D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon.''

Date of issuance: December 1, 2000.

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

Amendment No.: 95.

Facility Operating License No. NPF–69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** August 23, 2000 (65 FR 51358).

The November 3, 2000, submittal did not change the initial proposed no significant hazards consideration determination.

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated December 4, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 11 and May 12, 2000, as supplemented by letters dated June 13, June 16, July 14, September 21, October 26, and November 3, 2000.

Brief description of amendment: The amendment grants a conforming amendment to the License and the Technical Specifications for the approval of the transfer of the license for the Indian Point Nuclear Generating Unit No. 3 (IP3) held by the Power Authority of the State of New York to Entergy Nuclear IP3, LLC. to possess and use IP3 and to Entergy Nuclear Operations, Inc. (ENO) to possess, use and operate IP3.

Date of issuance: November 21, 2000.

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

Amendment No.: 203.

Facility Operating License No. DPR–64: Amendment revised the License and the Technical Specifications.

Date of initial notice in **Federal Register:** June 28, 2000 (65 FR 39954).

The supplemental information did not expand the scope of the application as originally noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 11 and May 12, 2000, as supplemented by letters dated June 13, June 16, July 14, September 21, October 26, and November 3, 2000.

Brief description of amendment: The amendment grants a conforming amendment to the License and the Technical Specifications for the approval of the transfer of the license for the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) held by the Power Authority of the State of New York to Entergy Nuclear FitzPatrick, LLC. to possess and use FitzPatrick and to Entergy Nuclear Operations, Inc. (ENO) to possess, use and operate FitzPatrick.

Date of issuance: November 21, 2000.

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

Amendment No.: 268.

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

Date of initial notice in **Federal Register:** June 28, 2000 (65 FR 39953).

The supplemental information did not expand the scope of the application as originally noticed in the **Federal Register**.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 9, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 7, 2000, as supplemented on August 9 and October 12, 2000.

Brief description of amendments: The amendments modify the Salem Unit Nos. 1 and 2 Technical Specifications (TS), and revise surveillance requirements associated with Auxiliary Feedwater (AFW) Pump testing described in TS 4.7.1.2.b by replacing the current wording with that of improved Standard TSs, NUREG–1431, ''Standard Technical Specifications, Westinghouse Plants.''

Date of issuance: December 5, 2000.

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

Amendment Nos.: 238 and 219.

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

Date of initial notice in **Federal Register:** June 14, 2000 (65 FR 37428).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 5, 2000.

No significant hazards consideration comments received: No.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: March 8, 2000, as supplemented April 5, 2000, and October 25, 2000.

Brief description of amendment: The amendment revises the Technical Specifications through revision to the storage configuration requirements within the existing storage racks and taking credit for a limited amount of soluble boron.

Date of issuance: December 7, 2000.

Effective date: December 7, 2000.

Amendment No.: 79.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 5, 2000 (65 FR 17918).

The April 5, 2000, and October 25, 2000, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–390 and 50–391, Watts Bar Nuclear Plant Units 1 and 2, Rhea County, Tennessee

Date of application for amendment: March 10, 2000, as supplemented November 6 and 9, 2000 and November 21, 2000 (two letters).

Brief description of amendment: Changed the Operating License to incorporate Physical Security/Contingency Plan—Tamper Indicating/Line Supervision Alarms Testing Frequency at Watts Bar Nuclear Plant (WBN) Units 1 and 2.

Date of issuance: December 5, 2000.

Effective date: December 5, 2000.

Amendment No.: 29 and 29.

Facility Operating License No. NPF–90: Amendment revises the Operating License.

Date of initial notice in **Federal Register:** September 20, 2000 (65 FR 56957). The November 6, 9, and 21, 2000, supplements provided clarifying information that did not change the scope of the initial proposed no significant hazards consideration determination.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of application for amendment: October 30, 2000, as supplemented November 15 and 22, 2000.

Brief description of amendment: Allow a one-time-only increase in the diesel generator Action Completion Time from 72 hours to 10 days to facilitate repairs to an emergency diesel generator to improve reliability.

Date of issuance: December 8, 2000.

Effective date: December 8, 2000.

Amendment No.: 30.

Facility Operating License No. NPF–90: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** November 3, 2000 (65 FR 66266). The November 15 and 22, 2000 supplements provided clarifying information that did not change the scope of the initial proposed no significant hazards consideration determimination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 8, 2000.

No significant hazards consideration comments received: No.

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: August 10, 2000.

Brief description of amendments: The amendments change Technical Specification (TS) 5.6.5, ''Core Operating Limits Report,'' to incorporate the latest, Nuclear Regulatory Commission (NRC)-approved methodology for analysis of large break loss-of-coolant accidents (LBLOCAs) for Comanche Peak Steam Electric Station, Units 1 and 2. The acceptability of this change to TS 5.6.5 is based upon the NRC staff's conclusion that the LBLOCA analysis methodology described in TXU Electric's Topical Report ERX–2000–002–P, ''Revised Large Break Loss of Coolant Accident Methodology,'' March 2000, is acceptable, as addressed in the associated Safety Evaluation.

Date of issuance: October 6, 2000.

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

Amendment Nos.: 80 and 80.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 23, 2000 (65 FR 51363).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: May 2, 2000, as supplemented August 30, 2000.

Brief description of amendments: The amendments change the CPSES Security Plan to: (1) Allow response team members to perform compensatory measures for protective area intrusion detection or closed circuit television failure, and (2) to modify the patrol frequency for the protected area.

Date of issuance: December 5, 2000.

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

Amendment Nos.: 82 and 82.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Security Plan.

Date of initial notice in **Federal Register:** October 4, 2000 (65 FR 59226).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 5, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 21st day of December 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

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NUCLEAR WASTE TECHNICAL REVIEW BOARD

Board Meeting: January 30–31, 2001—Amargosa Valley, Nevada: Discussions of the Status of DOE Studies Related to a Potential Yucca Mountain, Nevada, Repository for Spent Nuclear Fuel and High-Level Radioactive Waste; Update on Scientific and Engineering Studies Undertaken at the Yucca Mountain site; and Update on the DOE's Development of a Safety Strategy for a Potential Yucca Mountain Repository

Pursuant to its authority under section 5051 of Public Law 100–203, Nuclear Waste Policy Amendments Act of 1987, on Tuesday and Wednesday, January 30 and January 31, 2001, the Nuclear Waste Technical Review Board (Board) will be in Amargosa Valley, Nevada, to discuss U.S. Department of Energy (DOE) efforts to characterize a site at Yucca Mountain, Nevada, as the possible location of a permanent repository for spent nuclear fuel and high-level radioactive waste. The Board will ask the DOE to address several questions about important technical and scientific issues related to evaluating the suitability of the potential repository site. The meeting is open to the public, and several opportunities for public comment will be provided. The Board is charged by Congress with reviewing the technical and scientific validity of DOE activities related to civilian radioactive waste management.

The Board meeting will be held at the Longstreet Inn, HCR 70, Box 559, Amargosa Valley, Nevada. The telephone number is (775) 372–1777; the fax number is (775) 372–1280. The meeting will start at 8:00 a.m. on both days and will be open to the public.

Representatives of Nye County will lead off the meeting on Tuesday, January 30, with a greeting, which will be followed by the introduction of the Acting Director of the DOE's Office of Civilian Radioactive Waste Management and the General Manager of the new contractor for the Yucca Mountain Project, Bechtel SAIC Company LLC. The Board also will hear from Dr. Jean-Claude Duplessy, a member of France's National Scientific Evaluation Committee (CNE), which oversees the scientific and technical activities of the French nuclear waste disposal program. During the rest of the morning session, the DOE will make presentations on the status of the Yucca Mountain Project. It will give a general overview of the program and discuss plans for issuing the site recommendation consideration report. The DOE then will address a specific question from the Board dealing