approximately 3.7E6 Btu/hr. On December 5, 1997, with a decay heat rate of 0.7E6 Btu/hr and no SFP cooling, the licensee determined that it would take 72 hours for the SFP to heat up to 150 °F (66 °C) from an initial temperature of 80 °F (27 °C) . Since this determination, the decay heat rate has decreased by a factor of two to approximately 0.3E6 Btu/hr. Further, the evaporation rate of SFP water at 150 °F (66 °C) is approximately 11 gpm, well within the 30 gpm capacity of the SFP makeup water supplies.

The staff concludes that the licensee's request for an exemption from certain requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50 is acceptable in view of the greatly reduced offsite radiological consequences associated with the current plant status. The staff finds that the postulated dose to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the bounding accident, the length of time available gives confidence that mitigative actions and, if necessary, offsite measures for the public could be taken without preplanning. Therefore, the staff concludes that the requirement in 10 CFR 50.54(q) that emergency plans meet all the requirements of 10 CFR 50.47(b) and all the requirements of Appendix E to 10 CFR Part 50 is not now warranted at BRP, and an exemption from some of the onsite and offsite emergency planning standards and requirements is acceptable.

IV

The NRC staff has completed its review of the licensee's request for an exemption from the requirements of 10 CFR 50.54(q) that emergency plans must meet all of the standards of 10 CFR 50.47(b) and from the requirements of Appendix E to 10 CFR Part 50. This exemption includes partial exemption from the standards of 10 CFR 50.47(b)(3) through (7), and (9) and the requirements of 10 CFR Part 50, Appendix E, IV, ''Content of Emergency Plans;'' A.4; B; C; D.1 and 3; E.9.a and d; and F.1, 2, and 2.e. Further, this exemption covers all of the standards of 10 CFR 50.47(b)(10) and the requirements of 10 CFR Part 50, Appendix E, IV, A.3, 5, and 8; D.2; E.8 and 9.c; and F.2.c, d, and f. On the basis of its review, the NRC staff finds that the postulated dose to the general public from any reasonably conceivable accident would not exceed EPA PAGs and, for the bounding accident, the length of time available provides confidence that mitigative actions and, if necessary, offsite protective measures for the public could be taken without preplanning. The analyses submitted by the licensee are consistent with the statements made in its FHSR and proposed DEP, which state that any decommissioning activity will be bounded by the analyses presented therein and the considerations and assessments in the NRC's ''Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities'' (NUREG–0586). Consumers will continue to maintain and implement an onsite emergency preparedness organization capable of responding to and mitigating the consequences of radiological events still possible at the site and will continue to coordinate, as necessary, with offsite organizations to ensure effective emergency response to onsite situations, if needed. The staff finds the exemption from two requirements, 10 CFR 50.47(b)(9) and 10 CFR 50, Appendix E.IV.A.4, acceptable on the basis of the licensee's commitment to continue to maintain capabilities for dose assessment and personnel necessary to determine the potential impact of a radiological emergency on the general public. Thus, the underlying purpose of the regulations will not be adversely affected by eliminating offsite emergency planning activities and reducing the scope of onsite emergency planning.

For the foregoing reasons, the Commission has determined that, pursuant to 10 CFR 50.12, elimination of offsite emergency planning activities will not present undue risk to public health and safety, and is consistent with the common defense and security. Further, special circumstances are present as stated in 10 CFR 50.12(a)(2)(ii). Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (63 FR 50930).

This exemption is effective upon issuance.

Dated at Rockville, Maryland this 30th day of September 1998.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 98–26852 Filed 10–6–98; 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 Pub. L. 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 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 September 14, 1998, through September 25, 1998. The last biweekly notice was published on September 23, 1998 (63 FR 50932).

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

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration 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 November 6, 1998, 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 at the local public document room for the particular facility involved. 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of 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 at the local public document room for the particular facility involved.

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

Date of amendment request: October 28, 1997, as supplemented March 26, May 20, July 29, and August 13, 1998.

Description of amendment request: The proposed amendments would revise the current Technical Specifications (CTS) of each unit to conform with NUREG–1430, ''Standard Technical Specifications—Babcock and Wilcox Plants.'' The Commission had previously issued a Notice of Consideration of Issuance of Amendments published in the **Federal Register** on December 5, 1997 (62 FR 64405), covering all of the proposed Improved Technical Specification (ITS) changes that were within the scope of NUREG–1430 for the Oconee Nuclear Station. However, the submittals also contained proposed changes that are beyond the scope of NUREG–1430, which were not included in the staff's December 5, 1997, notice. The following descriptions and proposed no significant hazards analyses cover only the beyond-scope changes. Associated with each proposed change are administrative/editorial changes such that the new or revised requirements would fit into the format of NUREG–1430. Some changes are ''Less Restrictive'' (meaning that the new requirements being incorporated into the ITS are less restrictive than the CTS requirements) and some are ''More Restrictive.'' The basis for the no significant hazards determination is identical for all of the more restrictive items and is presented at the end of the following list of more restrictive beyond-scope items:

A. Certain NUREG and CTS Sections 3.1.3.5, 3.5.2.4.a, 3.5.2.5.b, 3.5.2.5.c, and 3.5.2.6, specify that they are applicable ''except during Mode 1 physics testing.'' The exception would not be included in the ITS and, therefore, the Mode 1 requirement would be applicable during the tests. The proposed change is more conservative since no exceptions would be allowed for physics tests conducted in Mode 1.

B. CTS 3.1.3.2 requires reactor coolant temperature to be greater than the criticality values of specified heatup limitation curves. This requirement would not be retained in the ITS. ITS 3.1.8, Limiting Condition for Operation (LCO) Part e, would be added to provide a restriction for loop average temperature to be greater than or equal to 520 °F when performing physics tests in Mode 2. ITS LCO 3.1.8 would permit suspending the requirements of ITS LCO 3.4.2, ''RCS (reactor coolant system) Minimum Temperature for Criticality,'' during physics tests initiated in Mode 2. Associated Actions and a surveillance requirement (SR) would be added to provide an appropriate required action when outside the limit and to verify operation within the limit periodically.

C. CTS Table 3.5.1–1 presently requires that the operator place the plant in hot shutdown (ITS equivalent of Mode 3) within 12 hours when the minimum channels Operable requirement is not met. The proposed change to the ITS would provide an equivalent requirement and add a requirement to open all control rod drive (CRD) trip breakers within 12 hours. ITS 3.3.3 Action B, and ITS 3.3.4 Action D, would be added to require that the unit be in Mode 3 in 12 hours with all CRD trip breakers open or that power be removed from all CRD trip breakers when the required action and associated completion time is not met in Mode 1, 2, or 3. For ITS 3.3.3, Action B would also apply when two or more reactor trip modules are inoperable in Mode 1, 2, or 3. The CTS presently requires entry into TS 3.0, which requires that the reactor be in hot shutdown (equivalent to ITS Mode 3) in 12 hours.

D. Note c would be added to ITS Table 3.3.8–1, Post Accident Monitoring Instrumentation, and referenced to Item No. 8, Containment Isolation Valve Position, to specify that position indication requirements apply only to the Containment Isolation Valves that are electrically controlled.

E. The applicability of Table 3.5.1–1 would be expanded to require wide range instruments to be operable in Mode 2, plus Modes 3, 4, and 5, with any control rod drive trip breaker in the closed position and the control rod drive system capable of rod withdrawal. In addition, a Note would define the upper limit of the applicable Modes for the required wide range instrument channels as being 10 percent indicated neutron power.

F. The applicability of ITS 3.3.14 would be expanded to include Mode 4 when the steam generator is relied upon for heat removal, which then would be consistent with the applicability of ITS LCO 3.7.5 for the emergency feedwater (EFW) system. ITS Specifications 3.3.14 and 3.3.15 would be added to address EFW system initiation circuitry and main steamline break and main feedwater isolation instrumentation separately. The specification titles, LCOs, actions, and SRs would be modified to reflect Oconee-specific terminology and design requirements. Where appropriate, ITS-required actions would be based on similar NUREG-required actions. EFW pump initiation circuitry operable requirement would be changed from 250 °F to greater than or equal to 246 °F.

G. ITS LCO 3.4.1, Departure from Nucleate Boiling Ratio (DNBR) Limits, are specified in the core operating limits report rather than in the LCO and SRs since they are subject to change with fuel cycle designs. The ITS LCO 3.4.1 actions would require restoring DNBR parameters to within limits within 2 hours or exiting the applicability for the specification within 12 additional hours. ITS SR 3.4.1.1, SR 3.4.1.2, and SR 3.4.1.3 would require verification that each DNBR parameter is within the limit at a 12-hour frequency. ITS SR 3.4.1.4 would require verification by measurement that total RCS flow is within limit at an 18-month frequency. Specification 3.4.1 would ensure that limits on RCS pressure, temperature, and flow rate are met to ensure that the core operates within the limits assumed for the plant safety analyses. These changes are more restrictive.

H. The NUREG allowed time to complete the SR after addition to core flood tank (CFT) of 6 hours would be changed to 12 hours. ITS SR 3.5.1.4 would require CFT boron concentration be sampled every 31 days or once within 12 hours after each solution volume increase greater than or equal to 80 gallons that is not the result of addition from a borated water source that meets CFT boron concentration requirements. Since the CTS does not specify the time limit following addition, the proposed ITS change is a more restrictive limit.

I. ITS 3.5.3 LCO Note 3 would be added to explicitly require that the low pressure injection (LPI) discharge header crossover valves be operable and capable of being opened manually when in Modes 1, 2, and 3. ITS 3.5.3 Action B would require that the LPI discharge header crossover valves be restored to operable status within 72 hours of being discovered incapable of being manually opened when in Modes 1, 2, and 3. ITS 3.5.3 Action D would require LCO 3.0.3 be entered immediately when one LPI train is inoperable in Modes 1, 2, and 3 concurrent with discovery that the LPI discharge header crossover valves are incapable of being opened manually in Modes 1, 2, and 3.

J. ITS 3.5.3 would require the LPI system to be operable in Modes 1, 2, 3, and 4. LCO Note 1 would be added to specify that only one LPI train is required to be operable in Mode 4. LCO Note 2 would be added to allow an LPI train to be considered operable during alignment, when aligned, or when operating if capable of being manually realigned to the LPI mode of operation. Action E would be added to require action be initiated immediately to restore the required LPI train to operable status and to require the reactor to be placed in Mode 5 within 24 hours when the required LPI train cannot be restored to OPERABLE status (provided a decay heat removal loop is available).

K. SR 3.9.4.1 would be modified to eliminate verification of a specific decay heat removal flow rate to verification every 12 hours that one decay heat removal loop is in operation.

L. Main feeder bus monitoring panel requirements and allowed outage time would be added to the ITS.

M. TS Section 3.7 would be revised to include the actual trip setpoint and/or allowable values for the loss of power sensing relays.

N. Battery performance discharge testing as related to battery operability would be added.

O. Battery charger testing, cell-to-cell resistance measurements, and battery discharge and overcharge conditions, surveillances would be added to ITS Section 3.8.

P. High Pressure Injection System discharge pressure allowable value in ITS Table 3.3.5–1 would be changed from 1500 pounds per square inch gauge (psig) to 1590 psig.

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 for the More Restrictive Items listed above, as follows:

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined that they do not represent a significant hazards consideration. The following is provided in support of this consideration.

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

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

For the less restrictive beyond-scope items, the basis for the no significant hazards consideration is unique for each item. The beyond-scope item and the licensee's basis supporting its determination that the proposed changes do not represent a significant hazards consideration follow:

A. A proposed change to the Note for ITS SR 3.1.4.3 would provide the additional flexibility for testing control rod drop times with reactor coolant flow conditions other than full flow, but with at least one reactor coolant pump (RCP) pump running. This would ensure that the testing is bounding by restricting operation of the unit to the RCP combination used during control rod drop testing and represents adoption of the NUREG rather than the CTS.

Basis for proposed no significant hazards consideration determination:

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

The control rods are used to support mitigation of the consequences of an accident; however, the control rod drop time variations are not considered the initiator of any previously analyzed accident. As such the proposed change in the method of performing the control rod drop time testing will not increase the probability of any accident previously evaluated. The proposed changes allow for testing the control rod drop times with less than a full complement of reactor coolant pumps operating. However, the operation of the plant is restricted to the pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, the drop times verified during testing will remain valid for mitigating the consequences of any accident previously evaluated. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure that the control rods are available for insertion of reactivity in the time frames consistent with the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided in the acceptable control rod drop times continues to be provided since these drop times have not been changed. The surveillance methodology is revised to allow testing with one, two, or three pumps operating. However, the operation of the plant is restricted to the reactor coolant pump combinations which maintain the margin of safety, i.e., those pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, this change does not involve a significant reduction in a margin of safety.

B. Required Action B.2.2 of ITS 3.3.11, 12, and 13, would be added to provide the option of closing the main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) in lieu of reducing main steam header pressure to less than 700 psig. Applicability would be changed to Modes 1 and 2, plus Mode 3 when the main steam header pressure is greater than 700 psig except when all MFCVs and SFCVs are closed.

Basis for proposed no significant hazards consideration determination:

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

The MSLB (main steamline break) and MFW (main feedwater) Isolation circuitry is not an initiator of analyzed events. Therefore, the probability of an accident is independent of the status of the MSLB and MFW Isolation circuitry. As such the proposed change does not involve a significant increase in the probability of an accident previously evaluated. The proposed change eliminates the requirement for MSLB and MFW Isolation circuitry OPERABILITY when all the MFCVs and SFCVs are closed. When the MFCVs and SFCVs are closed the MSLB and MFW Isolation circuitry has no safety function since its function is to close the MFCVs and SFCVs when conditions indicate [an] MSLB. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since MSLB and MFW Isolation circuitry requirements continue to require OPERABILITY when the reactor is in a condition that requires their function, the proposed change does not involve a significant reduction in a margin of safety.

C. ITS 3.3.15 Action A.1 would be added to allow 1 hour to declare the turbine stop valves (TSVs) inoperable prior to requiring that the unit shut down when one or more TSV closure channels is inoperable. ITS Specifications 3.3.14 and 3.3.15 would be added to address the emergency feedwater system initiation circuitry and main steamline break and main feedwater isolation instrumentation separately. The NUREG specification combines the emergency feedwater system initiation, main steamline isolation, and main feedwater isolation functions into one specification. The specification titles, LCOs, actions, and SRs would be modified to reflect Oconee-specific terminology and design requirements. Where appropriate, ITS-required actions would be based on similar NUREG-required actions.

Basis for proposed no significant hazards consideration determination:

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

This change establishes a 1 hour Completion Time during which the unit may continue operation with MSLB and MFW Isolation instrumentation inoperable. This change provides an opportunity to repair the inoperable instrumentation channel(s) prior to declaring the equipment supported by it inoperable. The addition of this allowed condition with a short Completion Time does not result in any hardware changes. The allowed condition also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Further, the consequences of an accident are the same during the additional one hour time period allowed for instrument channel restoration as it is during the time period currently allowed for restoring TSVs to OPERABLE status. Therefore, the change does not significantly increase the probability of occurrence of an accident previously evaluated.

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

The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change continues to ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The allowed condition has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, this new allowed condition does not involve a significant reduction in the margin of safety.

D. CTS 3.8.10 and 4.4.4.5 frequency would be changed from ''* * * immediately prior to refueling operation'' to ''Once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment'' in ITS SR 3.3.16.2 for testing frequency of the radiation monitor associated with the purge system valve isolation and ITS SR 3.9.3.2 for testing isolation function of the reactor building purge supply and exhaust valves.

Basis for proposed no significant hazards consideration determination:

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The isolation function of the radiation monitor associated with the purge system valves is not assumed to be an initiator of any analyzed event. As a result, the probability of an accident occurring is independent of the status of testing the isolation function of the radiation monitor associated with the purge system valves. This change eliminates the requirement for testing of this isolation function immediately prior to refueling operations. The change continues to require the isolation function to be OPERABLE and continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. This provides reasonable assurance the isolation function of the radiation monitor associated with the purge system valves remains OPERABLE. Therefore the consequence of an accident previously evaluated are not significantly increased.

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The isolation function of the Reactor Building Purge supply and exhaust valves is not assumed to be an initiator of any analyzed event. As a result, the probability of an accident occurring is independent of the status of testing the isolation function of the Reactor Building Purge supply and exhaust valves. This change eliminates the requirement for testing of the isolation function of the Reactor Building Purge supply and exhaust valves immediately prior to refueling operations. The change continues to require the isolation function of the Reactor Building Purge supply and exhaust valves train to be OPERABLE and continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. This continues to provide reasonable assurance the isolation function of the Reactor Building Purge supply and exhaust valves remains OPERABLE. Therefore the consequence of an accident previously evaluated are not significantly increased.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still require the isolation function of the radiation monitor associated with the purge system valves be OPERABLE. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still require the isolation function of the Reactor Building Purge supply and exhaust valves be OPERABLE. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The isolation function of the radiation monitor associated with the purge system valves is still required to be OPERABLE. This change continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. Therefore the margin of safety has not been significantly reduced.

The isolation function of the Reactor Building Purge supply and exhaust valves is still required to be OPERABLE. This change continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. Therefore the margin of safety has not been significantly reduced.

E. CTS 3.7.6 and 3.7.7 both require an inoperable voltage sensing relay to be restored within 72 hours. ITS 3.3.19 Required Action A.1 and ITS 3.3.20 Required Action A.1 would be incorporated to require that the inoperable channel be placed in trip within 72 hours. This change allows operation to continue indefinitely when the channel is placed in trip and continues to allow 72 hours to restore an inoperable channel that cannot be placed in trip.

Basis for proposed no significant hazards consideration determination:

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

This change allows indefinite continued operation with one voltage sensing channel inoperable, provided the inoperable voltage sensing channel is placed in trip within 72 hours. This action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel were to fail, the DGVP (degraded grid voltage protection) instrumentation can still perform its function. This change does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the DGVP instrumentation does not change (and therefore any initiation scenarios are not changed). Also, the change does not change the assumed response of the equipment in performing its specified function from that originally considered. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change ensures proper availability for the required DGVP function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety? This change to the DGVP instrumentation requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The change does not prevent the DGVP instrumentation from performing their function since the action places the DGVP instrumentation in a one-out-of-two condition for actuation versus the normal two-out-of-three logic. Thus, if another channel were to fail, the DGVP instrumentation could still perform its initiation functions. Therefore, this change to allow the DGVP initiation functions to operate indefinitely with one required DGVP instrument channel inoperable provided the channel is placed in the tripped condition within 72 hours, is not considered to involve a significant reduction in the margin of safety.

F. CTS Table 4.1–3 requires that CFT boron concentration be sampled monthly and after each makeup. ITS SR 3.5.1.4 requires it be sampled every 31 days and once within 12 hours after each solution increase greater than or equal to 80 gallons that is not the result of addition from a borated water source that meets CFT boron concentration requirements. Therefore, the ITS frequency is less restrictive than current requirements because sampling will be required once within 12 hours following the volume increase and source requirement. Also, the source of makeup would be changed from the ''borated water storage tank'' to ''a source that meets CFT boron concentration requirements.''

Basis for proposed no significant hazards consideration determination:

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

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate the tank inventory is within the required parameter limits. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The core flood tank boron concentration change resulting from volume addition from a source of known concentration is a readily calculated quantity and hence, a sample and analysis is not required to be assured of adequate boron concentration. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change continues to provide assurance of acceptable boron concentration since addition from a source of known concentration results in a readily identifiable resulting concentration. Therefore, a change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

G. The proposed change would specify actions to be taken for Borated Water Storage Tank (BWST) level, boron concentration, or temperature not being within specifications. Proposed ITS 3.5.4 Required Action C.1 would allow 12 hours to reach Mode 3 (i.e., an additional 6 hours over what is currently allowed by CTS 3.2.2) under such conditions.

Basis for proposed no significant hazards consideration determination:

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The time to be in MODE 3 is not assumed to be the initiator of any analyzed events. As a result, the probability of an analyzed event is independent of the time permitted to be in MODE 3. The consequences of an accident occurring during the 12 hours permitted to be in MODE 3 are no greater than the consequences of an accident occurring during the 6 hours currently permitted to place the unit in Hot Shutdown. Therefore, the probability and consequence of an accident previously evaluated are not significantly increased.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The time to place the unit in MODE 5 is appropriately limited. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time to place the unit in MODE 3 is not significantly greater than the time currently permitted to place the unit in Hot Shutdown and represents a reasonable time to accomplish the shutdown. Therefore, the extended time to place the unit in MODE 3 does not involve a significant reduction in the margin of safety.

H. CTS 3.3.4.b requires the BWST minimum boron concentration to be within the limit specified in the core operating limits report at a minimum temperature of 50 °F and would be changed to 45 °F in ITS SR 3.5.4.1. BWST maximum temperature would be changed from 100 °F to 115 °F.

Basis for proposed no significant hazards consideration determination:

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. BWST water temperature and volume are not assumed to be the initiators of any analyzed events. As a result, the probability of an analyzed event is independent of these values. The proposed change from allowable values based on the uncertainties associated with the instrument channel to an analytical limit for the parameter being measured continues to ensure that the limits on volume and pressure are maintained within analyzed values. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The analytical limits of variables established by the safety analysis have not been changed. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changing the limits from an allowable value based on the uncertainties associated with the instrument channel to an analytical limit for the parameter being measured does not involve a significant reduction in the margin of safety since the actual pressure and volume assumed in the safety analyses are not changed.

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

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW, Washington, DC.

NRC Project Director: Herbert N. Berkow.

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

Date of amendment request: July 31, 1998.

Description of amendment request: The proposed amendment would clarify requirements for diesel generator start voltage and frequency.

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

(1) Analyzed events are initiated by the failure of certain plant structures, systems or components. The proposed changes to the Clinton Power Stations (CPS) Technical Specifications revise the acceptance criteria for Surveillance Requirements (SRs) pertaining to the diesel generators (DGs). The DGs are not considered as initiators of any analyzed event. Thus, these changes do not increase the probability of any accident previously evaluated.

The consequences of analyzed events involving the diesel generators are dependent on the successful functioning of the diesel generator(s) to mitigate such events when a concurrent loss of offsite power is postulated. The proposed change in the acceptance criteria for testing of the DGs per the affected SRs accounts for DG governor performance in response to a fast start. Notwithstanding, the revised SRs will continue to ensure that minimum frequency and voltage are attained within the required time, thus satisfying permissive conditions required for closure of the DG output breaker. The SRs will also continue to ensure that proper steady-state voltage and frequency are attained consistent with proper DG governor and voltage regulator performance. Additionally, verification that permanently connected loads are energized within the required time (in response to a loss of offsite power or in response to a loss of coolant accident (LOCA) concurrent with a loss of offsite power) will continue to be performed pursuant to SRs not affected by the proposed changes. Thus, there is no impact on the capability of the DGs to perform their required safety function.

Based on the above, IP (Illinois Power Co.) has concluded that the proposed changes will not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated or in the set points that initiate protective or mitigative actions. As a result, no new failure modes are being introduced.

Additionally, there are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

Based on the above, IP has concluded that the proposed changes will not create the possibility of a new or different kind of accident not previously evaluated.

(3) As noted previously, the proposed changes to the acceptance criteria for testing of the DGs per the affected SRs accounts for the characteristics of the DG governor during a fast start, but they do not impact the effectiveness of such testing to provide assurance of DG operability. Thus, the proposed changes do not impact expected DG performance, including the capability for each DG to attain and maintain required voltage and frequency for accepting and supporting plant safety loads within the required time, as assumed in the plant safety analyses.

Margins of safety are established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of set points for the actuation of equipment relied upon to respond to an event. With respect to any margins of safety associated with the diesel generators, and as noted previously, the proposed changes do not impact diesel generator performance. That is, the SRs as revised will continue to ensure that proper voltage and frequency are attained for closure of the DG output breaker, and for steady-state conditions consistent with proper DG governor and voltage regulator performance. In addition, the proposed changes involve no changes to any setpoints or settings associated with the diesel generators. On this basis, the proposed changes do not involve any changes to any assumptions of the plant safety analyses with regard to the function of the diesel generators. Thus, no margins of safety are impacted by the proposed changes.

Based on the above, IP has concluded that the proposed change will not result in 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.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, IL 62525.

NRC Project Director: Ronald R. Bellamy (acting).

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

Date of amendment request: August 17, 1998.

Description of amendment request: The proposed amendment would reduce the load at which the diesel generators are tested.

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) Analyzed events (or events bounded by analyzed events) are initiated by the failure of certain plant structures, systems or components. The scope of the proposed changes is limited only to the revision of several Surveillance Requirements (SRs) for testing of the standby emergency diesel generators (DGs). The DGs are not considered as initiators of any analyzed event. Thus, the proposed changes do not impact the probability of any accident previously evaluated.

The consequences of analyzed events are dependent on the successful functioning of credited equipment to mitigate such events. With respect to the proposed changes, there is no impact on the capability of credited equipment, i.e., the diesel generators, to perform as required (in the event of a loss of coolant accident concurrent with a loss of offsite power). Testing at reduced load levels reduces stress and wear on the diesel generators, while still ensuring that the DGs are adequately challenged at operating temperatures to confirm operability. In addition, reducing the minimum required load levels reduces time when, or the probability that, the short-term rating of any diesel generators is exceeded during testing. The resultant reduction in stress and wear increases DG availability.

Based on the above, IP (Illinois Power Co.) has concluded that the proposed changes will not result in a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated or in the set points that initiate protective or mitigative actions. As a result, no new failure modes are being introduced.

Based on the above, IP has concluded that the proposed changes will not create the possibility of a new or different kind of accident not previously evaluated.

(3) The revised Surveillance Requirements are consistent with the recommendations of RG [Regulatory Guide] 1.9, Revision 3. Testing at reduced load levels reduces stress and wear on the diesel generators, while still ensuring that the DGs are adequately challenged at operating temperatures to confirm operability. In addition, reducing the minimum required load levels reduces time when, or the probability that, the short-term rating of any diesel generators is exceeded during testing. The resultant reduction in stress and wear increases DG availability.

Margins of safety are established through the design of plant structures, systems and components, the parameters within which the plant is operated, and the establishment of set points for the actuation of equipment relied upon to respond to an event. With respect to any margins of safety associated with the diesel generators, the proposed changes do not impact diesel generator performance, and involve no changes to any setpoints or settings associated with the diesel generators, nor do the proposed changes involve any changes to any assumptions of the plant safety analyses with regard to the function of the diesel generators. Thus, no margins of safety are impacted by the proposed changes.

Based on the above, IP has concluded that the proposed changes will not result in 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.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, IL 61727.

Attorney for licensee: Leah Manning Stetzner, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, IL 62525.

NRC Project Director: Ronald R. Bellamy (Acting).

Indiana Michigan Power Company, Docket No. 50–315, Donald C. Cook Nuclear Plant, Unit 1, Berrien County, Michigan

Date of amendment request: August 28, 1998.

Description of amendment request: The proposed amendment would grant relief from the steam generator inspection surveillance requirement described in technical specification No. 4.4.5.3. The relief would allow the inspection to be deferred from April 8, 1999, until the next refueling outage for Donald C. Cook Nuclear Plant , Unit 1.

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

In accordance with CFR 50.92, the proposed amendment will not involve a significant hazards consideration if the changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed;
2. Create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated; or
3. Involve a significant reduction in a margin of safety.

Criterion 1

The last unit 1 surveillance was completed in the spring of 1997 and was the most thorough evaluation of the steam generators to date. Both standard and enhanced eddy current inspection techniques were employed to inspect the steam generator tubing. Additionally, a series of in situ pressure tests were performed to verify tubing integrity. Tube repairs consisting of hot leg tube end re-rolling and plugging were performed. Pre- and post-tube bundle pressure tests were conducted to verify the integrity of the repairs. A tube pull was also conducted to verify continued comformance with generic letter 95–05 requirements. The tube pull data did not identify any unexpected conditions or areas of concern. During the 1997 inspection, select secondary side visual and eddy current inspections were also performed to provide assurance of continued secondary side internals integrity.

Following the inspection, a condition monitoring and operational assessment, using data gathered during the steam generator inspections and tests, was made to determine whether steam generator leakage and structural integrity could be maintained throughout the upcoming cycle (cycle 16).

The unit was subsequently restarted and the steam generators operated without incident when a unit shutdown occurred in September of 1997.

Throughout the cycle 16 operating period, a relatively low reactor coolant temperature was maintained. By maintaining a T-hot temperature of approximately 586 °F during the operating period, corrosion impact on the steam generator tubes was minimized.

Throughout the operating period, steam generator primary-to-secondary leakrate monitoring was performed to assure conformance with T/S requirements. Historically, Unit 1 has not experienced a forced shutdown because of leakrate concerns.

During the shutdown period, the steam generators have been maintained under lay-up conditions, which comply with or exceed the industry standard practice. These practices are designed to mitigate the corrosive environment within the steam generators.

The previous cycle 16 integrity assessment has been re-visited to provide reasonable assurance conclusions made remain valid given the extended shutdown period. This re-assessment considered the initial cycle runtime, the shutdown period and subsequent operation through the end of the current fuel cycle. These results confirm the findings of the initial evaluation (i.e., that adequate steam generator integrity will be maintained throughout the current cycle).

The proposed change will not affect the scope, methodology, acceptance limit, or corrective measures of the existing steam generator examination program. As adequate integrity will be maintained, the probability and consequences of an accident previously analyzed due to leaking or degraded tubes is not increased by the proposed change.

Criterion 2

We have determined that this extension will not result in a change in plant configuration or operation. Plant systems and components will not be operated in a different manner as a result of this change. No plant modifications or changes in methods of operation will result from this change. Therefore, the extension will not create the possibility of a new or different kind of accident from what has been previously evaluated or analyzed.

Criterion 3

We have determined that the proposed extension request will not involve a significant reduction in a margin of safety. Re-assessment of the cycle 16 steam generator operational assessment report, which indicates structural and leakage integrity will be maintained throughout the cycle, has shown that the shutdown period will not adversely impact overall steam generator integrity.

This assessment concluded that when the reactor is shut down and the reactor coolant system is at a reduced temperature, the steam generators are not subject to conditions that lead to tube degradation. The actual number of days that the steam generators will be subjected to an environment conducive to tube degradation is not being increased under this request. Therefore, this request is judged not to 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.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

NRC Project Director: Ronald R. Bellamy (Acting).

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: August 12, 1998.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) by updating the list of documents specified in TS 6.9.1.8b that describe the analytical methods used to determine the core operating limits. The changes can be categorized as: (1) The analysis methodology is unchanged, but the reference has been clarified by identifying the specific revision, supplements, and dates for the revision; (2) the analysis methodology is unchanged and the reference is being added for completeness and; (3) the analysis methodology is being changed. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed change in reference 4 of Technical Specification Section 6.9.1.8b revises the steam line break analysis methodology to be applied to Millstone Unit No. 2 and clarifies the references to the Siemens topical reports. The other changes are clarifications or additions for completeness and do not represent a change in the approved methodology for Millstone Unit No. 2. The change in methodology is associated with the interference between XTGPWR, the neutronics code, and XCOBRA–IIIC, the thermal hydraulics code. It has no impact on plant equipment operation. Since the change only affects the analysis of the events, it cannot affect the likelihood or consequences of these events. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The sentence on page 6–19, starting with ''The acceptable Millstone 2 * * *.'' and ending with ''* * * dated October, 1988,'' references the document ANF–88–126, ''Millstone Unit 2 Cycle 10 Safety Analysis Report,'' which has been outdated because of the above mentioned changes in the methodology. The removal of this sentence is necessary to be consistent with methodology changes. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed change in reference 4 of Technical Specification Section 6.9.1.8b revises the steam line break analysis methodology to be applied to Millstone Unit No. 2 and clarifies the references to the Siemens topical reports. The other changes are clarifications or additions for completeness and do not represent a change in the approved methodology for Millstone Unit No. 2. The proposed change in reference 4 of Technical Specification Section 6.9.1.8b will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. It does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated.

The sentence on page 6–19, starting with ''The acceptable Millstone 2 * * *.'' and ending with ''* * * dated October, 1988,'' references an outdated document. The removal of this sentence is necessary to be consistent with methodology changes. The change does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated.

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

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

The proposed change in reference 4 of Technical Specification Section 6.9.1.8b revises the steam line break analysis methodology to be applied to Millstone Unit No. 2 and clarifies the references to the Siemens topical reports. The other changes are clarifications or additions for completeness and do not represent a change in the approved methodology for Millstone Unit No. 2. The change in steam line break methodology is associated with the interface between XTGPWR, the neutronics code, and XCOBRA–IIIC, the thermal hydraulics code. The change will result in a better correlation between the two computer codes, which is the intent of the iteration process. This will result in more accurate results while still maintaining a conservative modeling of the event. The most significant impact is on the low RCS [reactor coolant system] flow cases associated with loss of offsite power. These cases are not limiting when compared to the offsite power available cases. The improved references will clearly identify the approved Siemens Topical Reports applicable to Millstone Unit No. 2 and will ensure that methodology changes will be identified and submitted to the NRC for approval as required. The sentence on page 6–19, starting with ''The acceptable Millstone 2 * * *.'' and ending with ''* * * dated October, 1988,'' references an outdated document. The removal of this sentence is necessary to be consistent with methodology changes.

Therefore, the proposed changes will not result in a significant reduction in the margin of safety as defined in the Bases for Technical Specifications covered in this License Amendment Request.

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.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, PO Box 270, Hartford, Connecticut.

NRC Project Director: William M. Dean.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of amendment request: August 15, 1996, as supplemented March 19, 1998.

Description of amendment request: The proposed amendment revises the Technical Specifications so that either 8 or 12 hour shifts will be considered ''normal'' and 40 hours will be considered a ''nominal'' week, changes the wording for surveillances required ''once per shift'' to ''once per 12 hours,'' clarifies the ''once per hour'' wording related to fire watch patrols, and makes a number of other typographical corrections and clarifications.

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

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

(With respect to shift definition and editorial changes:) This change does not affect the physical configuration of the plant or how it is operated, as such, it is not the initiator of any plant event. Working a ''normal'' 12-hour shift is no different from working a ''normal'' 8-hour shift with 4-hours of overtime which has been an accepted and approved practice for years. Therefore, the proposed changes will not result in any increase in the probability of an accident occurring. The intent is still that operators will not work excessive overtime either on a daily, or weekly basis.

The typographical errors, clarifications and title changes do not involve technical issues and as such do not involve safety issues, and therefore do not effect [sic] the chances or consequences of an accident.

(With respect to surveillance and fire watch patrol interval:) This change does not affect the physical configuration of the plant or how it is operated. As such, it is not the initiator of any plant event. This change clarifies the intervals in which Sensor Checks, Surveillances, and fire watch patrols must be completed. As described above [in the supplement], the 12-hour interval has been determined acceptable for the specified Sensor Checks and Surveillances based on Monticello and industry experience which demonstrates instrumentation and channel failures are rare. This change conforms the Monticello TS (Technical Specifications) to NUREG–1433 and clarifies the intervals in which checks must be completed.

Completing fire watch patrols on a one hour +25% interval will require patrols on an hourly basis, while providing flexibility to complete the patrols within a 15 minute window. In addition to the Technical Specification required fire watches, additional individuals are often in the plant proper, so the required hourly fire watch patrols are only part of the entire program for fire detection.

Therefore, the proposed changes will not result in a significant increase in the probability of an accident occurring.

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

(With respect to shift definition and editorial changes:) This change does not affect the physical configuration of the plant or how it is operated. Therefore, revising the length of a ''normal'' shift or correcting minor errors does not create the possibility of a new or different kind of accident from any previously evaluated. As such, it is not the initiator of any plant event.

(With respect to surveillance and fire watch patrol interval:) Revising the wording to ''once per 12 hours'' or ''once per hour (+25%)'' does not create the possibility of a new or different kind of accident from any previously evaluated. No new or different surveillance activities are proposed, nor are any being deleted. As such, it is not the initiator of any plant event.

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

(With respect to shift definition and editorial changes:) This change does not affect the physical configuration of the plant or how it is operated. The level of expertise on shift will not be diminished or changed as a result of this change. Therefore, this change will not reduce the margin of safety.

(With respect to surveillance and fire watch patrol interval:) This change does not affect the physical configuration of the plant or how it is operated. The level of expertise on shift will not be diminished or changed, nor will it reduce the functionality of plant equipment. This change requires Sensor Checks, surveillances, and fire watch patrols be completed within industry guidelines.

The 12 hour interval has been determined acceptable based on industry experience which demonstrates channel failure is rare. The one hour interval for fire watch patrols has also been an accepted industry standard. In addition to the Technical Specification required fire watches, additional individuals are often in the plant proper, so the required hourly fire watch patrols are only part of the entire program for fire detection. The proposed change simply defines the acceptable interval during which the task must be performed. Therefore, this change does not constitute 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.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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

NRC Project Director: Cynthia A. Carpenter.

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 request: January 14, 1998, as supplemented by letter dated May 19, 1998.

Description of amendment request: The proposed amendment would approve a modification to the Diablo Canyon Power Plant, Unit Nos. 1 and 2, 230 kV transmission system. The modifications include installation of new 230/12kV startup transformers with automatic load tap changers, along with installation of shunt capacitor banks. The transformers will assure that voltage on the plant 12 kV and 4 kV buses is maintained within limits, while the capacitor banks assure adequate VAR support.

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 replacement of the startup transformers (SUTs) with new transformers equipped with load tap changers (LTCs) for voltage control does not alter the original configuration of the electrical distribution system and hence, will not increase the probability of occurrence of an accident previously evaluated.

The replacement of the SUTs with new transformers equipped with LTCs will enhance the capability of the 12 kV and 4 kV electrical distribution systems to maintain sufficient voltage for successful transfer of the plant auxiliary loads to the startup source following a unit trip. This change eliminates the potential for ''double sequencing'' (starting loads from the 230 kV system, subsequent voltage degradation causes load shedding and restarting from the diesel generators) of the 4 kV vital loads during an accident by providing adequate voltage to the 4 kV vital buses from the 230 kV source. The maintenance of adequate voltage at the 4 kV vital buses prevents the second level undervoltage relay (SLUR) action. The LTC will automatically maintain adequate voltage at the terminals of the vital equipment under design basis accident conditions. Therefore, engineered safety feature equipment will function as previously evaluated.

The manual operation of the Unit 2 LTC while in a standby mode will not increase the probability of an accident since normally none of the plant loads are energized from the 230 kV system. Plant loads are only powered from the 230 kV system during short periods of unit startup and shutdown. Loss of the 230 kV system while the operating plant loads are fed from the 25/500 kV system cannot initiate an accident since the system is not connected to plant equipment if the loads are supplied by the 25/500 kV system. Therefore, the proposed modifications will not increase the probability of an accident previously evaluated. The manual operation of the Unit 2 LTC assures adequate voltage is supplied to Unit 2 safety equipment in the event of an accident. Therefore, the proposed modification will not increase the consequences of an accident.

The installation of the shunt capacitors at the Diablo Canyon Power Plant switchyard and Mesa Substation to replace the VAR support from Morro Bay Power Plant (MBPP), assuming no MBPP generation, does not alter the capability or availability of the offsite power source. Since shunt capacitors are considered more reliable than generators, it adds to the reliability of the 230 kV system and will not increase the probability of an accident previously evaluated.

Even if 230 kV voltage were lost or became degraded, the first or second level undervoltage relays will initiate transfer to the diesel generators should there be a loss or degraded 230 kV system while feeding the vital loads from the 230 kV system. This scenario is evaluated in Final Safety Analysis Report (FSAR) Update Section 15.2.9.1 ''Loss of Offsite Power to the Station Auxiliaries.''

Therefore, the changes will not increase the consequences of an accident previously evaluated since the safety-related loads will function as required.

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

The change does not result in a change in operation, maintenance, physical change, or procedural change that could create the possibility of an accident that is of a new or different type than previously evaluated.

The replacement SUTs and the installation of the shunt capacitors to replace MBPP serves the same function as the original design and do not create the possibility of a new or different type of accident. Should there be a loss of offsite power, the onsite power source (diesel generators) will provide power to the loads. The FSAR already includes an evaluation for station blackout if there is a total loss of both onsite and offsite power.

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

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

The replacement transformers and the installation of the shunt capacitors will not cause a reduction in the margin of safety as defined in the basis for any Technical Specification (TS). The minimum voltage required for safe shutdown is defined in TS Table 3.3.4, Functional Unit 7.b, ''Second Level Undervoltage Relay (SLUR) setting.'' By replacing the existing SUTs with automatic LTC transformers, the vital 4 kV bus voltage will be automatically maintained at a sufficiently higher value during normal operation such that during an accident, the minimum 4 kV vital bus voltages after the bus transfer will be adequate to prevent SLUR actuation. The installation of the shunt capacitors will assure adequate VAR support that was previously provided by operation of the MBPP in the Los Padres Region of PG&E's service territory for present peak load and future peak load growth under worse case line outage conditions.

During the interim period between January and February 1998, when manual control of the Unit 2 SUT LTC will be utilized to maintain adequate voltage at the 12 kV and 4 kV buses, the margin of safety is not reduced since the adjustment of the LTC will assure stable voltage for the vital buses.

Therefore, there is no reduction in a margin of safety as defined in the basis for any TS.

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 requests involve no significant hazards consideration.

Local Public Document Room Location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

NRC Project Director: William H. Bateman.

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 request: March 18, 1998.

Description of amendment request: The proposed amendment would approve a change in the way passive failures in the auxiliary saltwater (ASW) and component cooling water (CCW) systems are mitigated during the long-term recovery period following a loss-of-coolant accident (LOCA). Specifically, plant procedures would no longer require ASW and CCW system train separation after the transfer to hot leg recirculation following a LOCA.

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 changes revise the way passive failures are mitigated in the auxiliary saltwater (ASW) and component cooling water (CCW) systems. Specifically, plant procedures would no longer require ASW and CCW train separation after transfer to hot leg recirculation following a loss-of-coolant accident. The decision to separate trains would be made by the Technical Support Center (TSC) after evaluation of plant conditions. Operation of the ASW and CCW systems during this period is required to mitigate the accident, therefore, the change in plant operation would not affect the probability of that accident occurring.

The change ensures the ASW and CCW systems will be able to mitigate an active or passive failure without the loss of safety function during the long-term (beginning 24 hours after the accident) period of recovery following an accident. Since the ASW and CCW systems will continue to perform their safety function, overall system performance is not affected, assumptions previously made in evaluating the consequences of the accident are not altered, and the consequences of the accident are not increased as a result of the change in plant operation.

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

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

The ASW and CCW systems function to mitigate the consequences of an accident. The change in operation ensures these system will be able to mitigate an active or passive failure without loss of safety function during the long-term (beginning 24 hours after the accident) period of recovery following an accident. Operation of the ASW and CCW systems in accordance with plant procedures, and the guidance on train separation provided to the TSC, ensure the design basis requirements for the ASW and CCW systems will continue to be met. Therefore, the ability of the ASW and CCW systems to mitigate the accident is not degraded. Required operator actions are similar to other operator actions specified in the FSAR that are considered acceptable by the NRC.

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

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

The change ensures the ASW and CCW systems will be able to mitigate an active or passive failure without loss of safety function during the long-term (beginning 24 hours after the accident) period of recovery following an accident. Since the ASW and CCW systems will continue to perform their safety function, there is no impact on any acceptance limits for ASW and CCW system operation assumed in the safety analysis, or on any Technical Specification (TS).

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

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 requests involve no significant hazards consideration.

Local Public Document Room Location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

NRC Project Director: William H. Bateman.

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 request: August 10, 1998,

Description of amendment request: The proposed amendments would revise the combined Technical Specifications (TS) for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 to revise TS 3/4.3.2, Table 3.3–5, ''Engineered Safety Features Response Times,'' to add the response times for closure of the main feedwater regulating valves (MFRVs) and MFRV bypass valves, and trip of the main feedwater pumps (MFWPs). The change would also revise TS 3/4.7.1.7 to add a limiting condition for operation (LCO), actions, and surveillance requirements for the MFWP turbine stop valves, and would revise the actions and surveillance requirements for the MFRVs, MFRV bypass valves, and main feedwater isolation valves (MFIVs) to be consistent with the NUREG–1431 requirements. Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The proposed changes to the Technical Specifications (TS) to add response time requirements for the main feedwater regulating valve (MFRV) and associated bypass valves and the main feedwater pump (MFWP) trip provide more restrictive TS requirements that are consistent with current plant practice. They do not change the function or operation of any plant equipment or affect the response of that equipment if it is called upon to operate. These more restrictive requirements are imposed to ensure the affected components are maintained consistent with the safety analyses and licensing bases.

The proposed changes to: (1) Revise the actions to apply to one or more main feedwater isolation valves (MFIVs), and MFRVs and associated bypass valves, (2) extend the action completion time from 4 hours to 72 hours, (3) provide actions when two valves affecting the feedwater isolation capability for a flow path are inoperable, (4) add actions for an inoperable MFWP turbine stop valve, and (5) allow separate action entry for each inoperable valve unless the feedwater isolation capability for a flow path is affected, do not change the function or operation of any plant equipment or affect the response of that equipment if it is called on to operate. The actions account for the redundancy provided by the remaining valves and the MFWP trip, and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow path. A probabilistic risk assessment, performed to assess the increase in annual core damage frequency (CDF) associated with the increase in allowable outage time, determined the increase in annual CDF to be approximately 1.5 percent. That increase in annual CDF is considered non-risk significant per the Electric Power Research Institute ''PSA Application Guide.''

The addition of the limiting condition for operation, actions, and surveillance requirements for the MFWP turbine stop valves, and the addition of the surveillance requirement for the MFIVs, MFRVs, and MFRV bypass valves are more restrictive requirements that ensure these components are operable and capable of performing their safety function. They do not change the function or operation of any plant equipment or affect the response of that equipment if it is called on to operate. The proposed surveillance intervals are supported by the operating, maintenance, and surveillance histories of the valves.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the parameters governing normal plant operation. The changes imposed are consistent with the assumptions made in the accident analyses and licensing basis.

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

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

The proposed changes to the TS impose requirements consistent with the assumptions in the safety analyses and current licensing bases, and reflect current plant practice. They do not alter the margins of safety established in previous accident and transient analysis.

Therefore, none of the proposed changes involves 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 requests involve no significant hazards consideration.

Local Public Document Room Location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

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

NRC Project Director: William H. Bateman.

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

Date of amendment request: September 8, 1998.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Appendix C, ''Additional Conditions,'' to authorize the use of non-Class 1E single cell battery chargers, with proper electrical isolation, for charging connected cells in OPERABLE Class 1E batteries. The single cell chargers would be used to restore individual cell float voltage to the normal limit specified in TS Table 4.8.2.1–1.

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

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

The proposed change permits the use of an industry accepted method to restore a battery cell to its design basis from an OPERABLE but degraded condition or to prevent a cell from becoming degraded. IEEE Std 450–1995, ''IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead Storage Batteries for Stationary Applications,'' states that single cell charging is an acceptable method of correcting low cell voltage or low specific gravity conditions for a single cell or for a small number of cells.

At least two class 1E fuses in series will be used on both the positive and negative leads between the battery and the charger to protect the battery if a fault should develop in the charger. The battery charger design includes diodes, a power transformer and control circuitry to prevent draining the connected cells in the event of a short circuit in the 120 Volt ac source or a loss of charger input or output voltage. Charger output is controlled automatically to prevent overcharging the connected cells.

In the event of a controller failure resulting in charger overvoltage, procedural controls governing the use of the charger ensure the condition is detected and corrected before failure of a connected cell occurs. While the single cell charger is connected, procedures will require periodic checks to verify proper charger operation and to measure electrolyte level, temperature and specific gravity for the cells being charged. Monitoring will be performed at least once every eight hours, a frequency sufficient to ensure compliance with the ACTION requirements of Technical Specification 3.8.2.1.

An insulating material will be used to minimize the possibility of shorting leads or clips at the battery. Administrative controls governing the use and storage of transient loads are sufficient to ensure the use of single cell battery chargers does not create a potential missile hazard to safety related systems, structures and components.

The Class 1E dc system is not an accident initiator. It supports the operation of safety related equipment required for the safe shutdown of the plant and for the mitigation of accident conditions. Therefore, the proposed change does not increase the probability of an accident previously evaluated.

The station's dc systems will be operable to mitigate the consequences of an accident previously evaluated. Single cell charging would be limited to one OPERABLE class 1E battery bank at a time. Therefore, failure of a class 1E battery as a result of single cell charging would be limited to a single channel and would not reduce the number of OPERABLE dc sources below that required to safely shutdown the plant. Administrative controls would also prohibit the use of single cell charging for an OPERABLE class 1E battery if less than the minimum number of class 1E batteries required by Technical Specifications are OPERABLE.

The proposed change does not cause the capability of the class 1E dc system to be degraded below the level assumed for any accident described in the (safety analysis report) SAR. It would enhance the availability of safety related equipment required for the safe shutdown of the plant and for the mitigation of accident conditions. Therefore the radiological consequences of an accident will remain inside the design basis while single cell charging is performed on an OPERABLE battery.

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

The potential to adversely affect the Class 1E batteries is minimized by the use of Class 1E fuses and by appropriate administrative controls. Failure modes associated with the proposed change are bounded by the loss of a Class 1E battery bank which was previously evaluated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change permits the use of non-Class 1E single cell battery chargers, with proper electrical isolation, for charging connected cells in OPERABLE class 1E batteries. This would allow parameters for an individual cell or for a small number of cells to be restored to the normal values specified in Technical Specifications without affecting the remainder of the cells in the battery. Increased cell monitoring after single cell charging, together with PSE&G's corrective action program which requires degraded and non-conforming conditions to be documented and evaluated, provides assurance that the use of single cell charging will not cause long-term cell degradation to go undetected. Since all battery cells are required to be maintained within the allowable values specified in Technical Specifications, and since the use of the single cell charger will not adversely affect battery capacity or capability, the proposed change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, NJ 08070.

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

NRC Project Director: Robert A. Capra.

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1

Fairfield County, South Carolina.

Date of amendment request: July 1, 1998.

Description of amendment request: The proposed amendment would revise Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Surveillance Requirement 4.7.7.e to remove the ''during shutdown'' condition from the specified test interval. Removing the ''during shutdown'' wording from the TS would allow VCSNS to perform on-line snubber testing, and would make the up to 25 percent allowable interval extension in Surveillance Requirement 4.0.2 apply to the specified snubber surveillance interval. The proposed amendment would also make administrative changes to Surveillance Requirement 4.7.7.g and BASES 3/4.2.2 and 3/4.2.3 to correct typographical errors.

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 probability or consequences of an accident previously evaluated is not significantly increased.

The proposed change will not affect system operation or performance, nor do they affect any Engineered Safety Features actuation setpoints or accident mitigation capabilities. NUREG/CR–6027 supports the determination that piping failure due to a snubber single failure is considered low. Therefore, the proposed changes will not significantly increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The changes to the situational testing requirements will not affect the method of operation of any system to which a snubber is attached. The proposed changes only address the plant mode at which a surveillance activity may be performed. No new or different accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of these changes. Therefore, the possibility of a new or different kind of accident other than those already evaluated will not be created by this change.

3. The margin of safety has not been significantly reduced.

This proposed change will not have an impact on the overall reliability of the snubber population. This is due, in part, to the fact that the snubber test plans are self correcting. As functional test failures are identified, additional snubbers are required to be tested. Thus, the reliability of the snubber population is maintained. The proposed change does not alter the intent or method by which the surveillances are conducted, does not involve any physical changes to the plant, does not alter the way any structure, system, or component functions, and does not modify the manner in which the plant is operated. Therefore the proposed change will not degrade the ability of the snubbers to perform their safety function or significantly decrease 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.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Acting Project Director: P. T. Kuo.

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

Date of amendment request: June 26, 1998, as supplemented by letter dated September 18, 1998.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) as follows: (1) The applicability of

Limiting Condition for Operation (LCO) 3.3.6 would be revised to refer to TS Tables 3.3.6–1 and 3.3.6–1; the TS Tables would be revised to add a column entitled ''APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS.'' Then, the applicable modes for Manual Initiation, Automatic Actuation Logic and Actuation Relays, and Safety Injection functions would be revised to include *only* Modes 1, 2, 3, and 4. Consistent with this proposed change, LCO 3.3.6, Condition C and Required Action C.2 would be revised to reflect that system level manual initiation and automatic actuation would not be required during core alterations and/or during movement of irradiated fuel assemblies within containment. Appropriate Bases changes are included to reflect the proposed changes; (2) LCO 3.9.4 would be revised to allow the equipment hatch and the emergency air locks to be open during core alterations and/or during movement of irradiated fuel assemblies within containment. In addition, the LCO statement would be revised to reflect that containment ventilation isolation (CVI) would be accomplished by manually closing the individual CVI valves as opposed to a system level manual or automatic initiation, consistent with the proposed changes to LCO 3.3.6. The surveillance requirements (SRs) would be revised to reflect the proposed change to the CVI and to reflect that the equipment hatch would be allowed to be open. Appropriate Bases changes are included to reflect the proposed changes; (3) LCO 3.7.6a, ''Condensate Storage Tank (CST)—(Non-redundant CSTs),'' would be deleted. This LCO was created to address a design condition that rendered the CSTs nonredundant. A note was added stating that this LCO was only applicable to the unit(s) that have not completed design modifications required for redundant CSTs and that the LCO would no longer be required when both units completed the design modifications. These design modifications have been completed; therefore, LCO 3.7.6a is no longer applicable, and LCO 3.7.6, ''Condensate Storage Tank (CST)—(Redundant CSTs),'' would be revised to delete the words ''(Redundant CSTs)'' from the title. Appropriate Bases changes are included to reflect the proposed changes.

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

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

No. The proposed changes would revise the VEGP [Vogtle Electric Generating Plant] Unit I and Unit 2 TS by removing requirements for automatic and system level manual containment ventilation isolation, and allow the emergency air lock and the equipment hatch to be open during core alterations and movement of Irradiated fuel assemblies inside containment. The containment penetrations affected by the proposed changes are not initiators for any accident previously evaluated. Allowing these penetrations to be open under the conditions specified will not affect the probability of any accident previously evaluated.

The existing VEGP TS allow the personnel air look doors to be open during core alterations and movement of irradiated fuel assemblies inside containment. The radiological consequences of a fuel handling accident inside containment have been determined to be below the Standard Review Plan (SRP) section 15.7.4 criteria and General Design Criteria (GDC) 19 criteria with the personnel air lock doors open. The proposed changes will not alter these previously determined consequences. The existing dose analysis bounds the proposed changes. Therefore, the proposed changes will not increase the consequences of any accident previously evaluated.

The proposed deletion of LCO 3.7.6a is an administrative change only. The requirements of LCO 3.7.6a applied only during the time that the condensate storage tanks (CSTs) were not redundant. Due to the implementation of design changes which make the CSTs redundant for each unit, the requirements of LCO 3.7.6a are no longer applicable. The CSTs (redundant or not) are not initiators for any accident previously evaluated. Now that the CSTs are redundant, the requirements of LCO 3.7.6a are no longer necessary to ensure the capability of the auxiliary feedwater system to perform its safety function. Therefore, the proposed deletion of LCO 3.7.6a will not affect the probability or consequences of any accident previously evaluated.

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

No. The proposed change does not create any new failure modes for any system or component, nor does it adversely affect plant operation. The previously determined radiological consequences of a fuel handling accident inside containment with the personnel air lock doors open remain bounding for operation under the proposed changes. No new single failure scenarios are created, and the proposed changes do not introduce any new challenges to components and systems that could result in a new or different kind of accident from any previously evaluated.

The proposed deletion of LCO 3.7.6a is an administrative change only. The requirements of LCO 3.7.6a applied only during the time that the condensate storage tanks (CSTs) were not redundant. Due to the implementation of design changes which make the CSTs redundant for each unit, the requirements of LCO 3.7.6a are no longer applicable. Now that the CSTs are redundant, the requirements of LCO 3.7.6a are no longer necessary to ensure the capability of the auxiliary feedwater system to perform its safety function. No new single failure scenarios are created, and the proposed changes do not introduce any new challenges to components and systems that could result in a new or different kind of accident from any previously evaluated. Therefore, the proposed deletion of LCO 3.7.6a will not create a new or different kind of accident from any accident previously evaluated.

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

No. The margin of safety for fission product release is 300 rem thyroid and 25 rem whole body as defined by 10 CFR (Part) 100. The previously determined radiological dose consequences for a fuel handling accident inside containment with the personnel air lock doors open remain bounding for operation under the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR (Part) 100 by virtue of the fact that they meet SRP Section 15.7.4 and GDC 19 acceptance criteria. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The proposed deletion of LCO 3.7.6a is an administrative change only. The requirements of LCO 3.7.6a applied only during the time that the condensate storage tanks (CSTs) were not redundant. Due to the implementation of design changes which make the CSTs redundant for each unit, the requirements of LCO 3.7.6a are no longer applicable. Now that the CSTs are redundant, the requirements of LCO 3.7.6a are no longer necessary to ensure the capability of the auxiliary feedwater system to perform its safety function. Therefore, LCO 3.7.6a is not necessary to maintain margin of safety and the proposed change will 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.

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: July 13, 1998

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 1.1 Definitions for ''Engineered Safety Feature (ESF) Response Time'' and ''Reactor Trip System (RTS) Response Time'' to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC. Changes to the TS Bases have also been proposed.

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

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

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

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

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

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

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

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

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: September 3, 1998.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) to: (1) Support the replacement of the Nuclear Instrumentation System Source Range and Intermediate Range Channels and Post-Accident Neutron Flux Monitoring System; and (2) delete the requirement for performing response time testing of the source range channels and power range detector plateau voltage determinations.

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 power range low trip, the intermediate range trip, and the source range trip are designed to provide protection against power excursions during reactor startup or low-power operation. The source and intermediate range trips provide redundant protection during reactor startup or low-power operation. The changes to the source range and intermediate range instrumentation and setpoints, as well as the deletion of source range response time testing, do not affect any safety analysis conclusions because the source range and intermediate range trips are not explicitly credited in any design basis accident. Only the power range low trip setpoint is assumed to actuate to mitigate the uncontrolled rod cluster control assembly withdrawal accident. The high flux at shutdown alarm function during a boron dilution event will continue to be provided by the new source range detector system. No changes have been made to the setpoint assumed in the safety analyses. The new detector system is qualified in compliance with Regulatory Guide 1.97 and will also be used to provide post-accident monitoring. The functional and operability requirements for the power range channels are not affected by deleting the requirement for determining detector voltage plateaus.

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

2. The functional and operability requirements for the new detector system are the same as for the existing system as defined by the Technical Specifications. No credit is taken for the source and intermediate range trips in any of the design basis accidents. The high flux at shutdown alarm and post-accident monitoring functions continue to be met. The functional and operability requirements for the power range channels are not affected by deleting the requirement for determining detector voltage plateaus.

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

3. The functional and operability requirements for the new detector system are the same as for the existing system. The functional and operability requirements for the power range channels are not affected by deleting the requirement for determining detector voltage plateaus. The margin of safety provided by the previous Technical Specifications is not significantly affected because the proposed changes are based on the same accident analysis acceptance limits.

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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.

Local Public Document Room location: Burke County Public Library, 412 Fourth Street, Waynesboro, Georgia.

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

NRC Project Director: Herbert N. Berkow.

Tennessee Valley Authority, Docket No. 50–260 Browns Ferry Nuclear Plant Unit 2 Limestone County, Alabama

Date of amendment request: September 8, 1998.

Description of amendment request: The proposed amendment would revise the Browns Ferry Nuclear Plant (BFN) Unit 2 technical specifications (TS) to include provisions for enabling the Oscillation Power Range Monitor (OPRM) Upscale trip function in the Average Power Range Monitor (APRM). The APRM is part of the Power Range Neutron Monitoring (PRNM) system. The OPRM Upscale trip function provides protection from exceeding the fuel Minimum Critical Power Ratio (MCPR) safety limit in the event of thermal-hydraulic power oscillations, and thereby, provides compliance with Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria (GDC) 10 and 12.

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

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

The proposed amendment is to enable the OPRM Upscale trip function which is contained in the previously installed PRNM equipment. Enabling the OPRM hardware provides the long term stability solution required by Generic Letter 94–02.

This hardware incorporates the Option III detect and suppress solution reviewed and approved by the NRC in NEDO–31960, ''BWROG [Boiling Water Reactor Owners Group] Long Term Stability Solutions Licensing Methodology.'' The OPRM is designed to meet all requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic power oscillations prior to violating the fuel MCPR Safety Limit. The OPRM system provides this protection in the region of the power-to-flow map where instabilities can occur, including the region where ICAs (interim corrective actions) restricted operation because of stability concerns. Thus, the ICA restrictions on plant operations are deleted from the TS, including region avoidance and the requirement for the operator to manually scram the reactor with no recirculation loops operating. Operation at high core powers with low core flows may cause a slight, but not significant, increase in the probability that an instability can occur. This slight increase is acceptable because subsequent to the automatic detection of a design basis instability, the OPRM Upscale trip provides an automatic scram signal to the RPS [reactor protection system] which is faster protection than the operator-initiated manual scram required by the current ICAs. Because of this rapid automatic action, the consequences of an instability event are not increased as a result of the installation of the OPRM system because it eliminates dependence on operator actions.

Based on the above discussion, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment permits BFN to enable the OPRM power oscillation detect and suppress function provided in previously installed PRNM hardware, and it simultaneously deletes certain restrictions which preclude operation in regions of the power-to-flow map where oscillations potentially may occur. Enabling the OPRM Upscale trip function does not create any new system hardware interfaces nor create any new system interactions. Potential failures of the OPRM Upscale trip result either in failure to perform a mitigation action or in spurious initiation of a reactor scram. These failures would not create the possibility of a new or different kind of accident. Based on the above discussion, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously.

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

The OPRM Upscale trip function implements BWROG Stability Option III, which was developed to meet the requirements of GDC 10 and GDC 12 by providing a hardware system that detects the presence of thermal-hydraulic instabilities and automatically initiates the necessary actions to suppress the oscillations prior to violating the MCPR Safety Limit. The NRC has reviewed and accepted the Option III methodology described in Licensing Topical Report NEDO–31960 and concluded this solution will provide the intended protection. Therefore, it is concluded that there will be no reduction in the margin of safety as defined in TS as a result of enabling the OPRM Upscale trip function and simultaneously removing the operating restrictions previously imposed by the ICAs.

Based on the above discussion, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on its 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.

Local Public Document Room location: Athens Public Library, 405 E. South Street, Athens, Alabama 35611.

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

NRC Project Director: Frederick J. Hebdon.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: August 27, 1996, and as supplemented on July 22, 1998.

Description of amendment request: The amendment request removes the Technical Specification requirements for the Main Steam Isolation Valve Leakage Control System, and increases the allowable leak rate specified for the main steam lines. The Perry facility is a pilot plant in the collaborative efforts of the Nuclear Regulatory Commission, the Nuclear Energy Institute, and the Electric Power Research Institute for implementation of the NRC research documented in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants.'' The proposed changes are based on reanalysis of the design basis Loss of Coolant Accident using the revised accident source term from NUREG–1465 and the NEI document entitled ''Generic Framework for Application of Revised Accident Source Term to Operating Plants.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration 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 removes the Technical Specification requirements for the Main Steam Isolation Valve Leakage Control System (MSIV–LCS), and increases the allowable leak rate specified for the main steam lines. Although the requirements for the MSIV–LCS are being removed (since credit is no longer taken for the system as part of the design basis accident analysis), OPERABILITY requirements on the Main Steam Shutoff Valves are being retained since the valves meet Criterion 3 of 10 CFR 50.36(c)(2)(ii). Removing the Technical Specification requirements of the MSIV–LCS and increasing main steam line allowable leakage rates has been addressed in the Loss of Coolant Accident (LOCA) reanalysis and does not adversely affect operation of other equipment or systems important to safety. These changes do not affect the precursors for accidents or transients analyzed in Chapter 15 of the Perry Nuclear Power Plant (PNPP) Updated Safety Analysis Report (USAR). Therefore, there is no increase in the probability of accidents previously evaluated.

The spectrum of LOCAs was considered to determine which would be most limiting with respect to radiological consequences. The worst case LOCA (i.e., main steam line break upstream of the inboard MSIV) off-site and Control Room doses have been reanalyzed using the revised design basis accident (DBA) source term (from NUREG–1465 and the Nuclear Energy Institute (NEI) document ''Generic Framework for Application of Revised Accident Source Term to Operating Plants'') in order to assess the radiological consequences of the increased main steam line leak rates, and not taking credit for the MSIV–LCS. The radiological analysis used conservative assumptions and analytical techniques. These conservatisms in the LOCA reanalysis have been determined to be comparable to the conservatisms utilized in the original analyses.

The results of the off-site and Control Room dose reanalysis are provided below.

DOSE RESULTS (REM)

		Proposed USAR dose*	Existing USAR dose	Regulatory limit **
Control Room	Whole Body	0.1	0.4	5
	Thyroid	16.2	29.2	30
	Skin	4.8	2.5	30
EAB	Whole Body	1.9	3.6	25
	Thyroid	157.9	140.8	300
LPZ	Whole Body	1.7	1.9	25
	Thyroid	130.3	144.7	300

* Rounded to nearest tenth.

** Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) dose limits are per 10 CFR 100.11. Control Room dose limits are per 10 CFR part 50 Appendix A, General Design Criterion (GDC) 19 and NUREG 0800 Standard Review Plan (SRP) Section 6.4.

As noted in the NEI Generic Framework Document (''Generic Framework for Application of Revised Accident Source Term to Operating Plants,'' EPRI TR-105909, Interim Report, November 1995), the acceptability of applications utilizing the revised accident source terms ''may be judged by the same licensing acceptance limits (e.g., dose limits in 10 CFR part 100) in use with the TID–14844 source term. That is, the licensee would show that the revised design basis, with either selective or essentially complete application of NUREG–1465 together with the plant changes under evaluation, results in doses no greater than these licensing acceptance limits.'' The off-site dose licensing acceptance limit for PNPP is 10 CFR part 100.11 (see Question 3 for details on the source of this PNPP licensing acceptance limit). The newly calculated radiological doses were lower for six of the seven factors evaluated. For the one factor which was higher, i.e., at the EAB for thyroid dose (from 140.8 REM to 157.9 REM), the dose remained significantly below the 10 CFR part 100 limit of 300 REM to the thyroid. This analysis demonstrated that the resulting off-site and Control Room doses were well below the regulatory limits contained in 10 CFR part 100, Reactor Site Criteria, and 10 CFR part 50, Appendix A, General Design Criterion (GDC) 19, Control Room. Therefore, the proposed changes do not involve a significant increase in the consequences of previously evaluated accidents.

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

The proposed change removes the Technical Specification requirements for the MSIV–LCS, retains the Technical Specification requirements for the Main Steam Shutoff Valves, and increases the allowable leak rate specified for the main steam lines.

Removing the Technical Specification requirements for the MSIV–LCS is based on reanalysis of off-site and Control Room doses, where the MSIV–LCS is not credited in the calculation. As noted above, the reanalysis utilizes the revised design basis accident (DBA) source terms. The limiting reanalysis case assumes that main steam line leakage is attenuated in the main steam line from the reactor vessel out to the outboard MSIV. This is the limiting scenario since the worst case single failure, and hence the most limiting analysis case, involves a failure to close the valve downstream of the outboard MSIV in each main steam line, i.e., the Main Steam Shutoff Valves (1N11F0020A,B,C AND D). Although this most limiting analysis case assumes a failure to close the Main Steam Shutoff Valves, retention of OPERABILITY requirements on these valves is appropriate to ensure the single failure analysis remains valid.

Not crediting the MSIV–LCS in the design basis accident analysis is consistent with the approach taken by several BWR licensees, which have applied for NRC approval of this change using an approach developed by the Boiling Water Reactor Owners Group (BWROG). The BWROG methodology involves seismically qualifying the main steam lines out to and including the non-safety related, non-seismic drain line and main condenser, and then using that volume to attenuate leakage past the MSIVs. At PNPP, the existence of safety related, seismically qualified piping leading to the safety related, Class 1E powered Main Steam Shutoff Valves (downstream of the outboard MSIV), together with the characteristics of the revised accident source term (i.e., predominantly aerosol which is largely retained in the drywell, containment and main steam lines) provides the option of taking credit only for the volume within the main steam lines for leakage attenuation.

Knowledge of the more physically correct source term timing and chemical form permits use of more appropriate mitigation techniques. Specifically, natural forces such as gravitational settling of aerosol (particulates) has been credited inside the drywell and in portions of the main steam lines, which significantly reduces the amount of radionuclides that could escape from the containment and into the environment. Also, based on a high radiation signal in the Control Room, the Containment Spray system would be operated post-LOCA for up to 24 hours (previous analyses assumed 6 hours of spray operation), in order to scrub released radionuclides from the containment atmosphere and into the suppression pool, and thus reduce the post-LOCA off-site and Control Room dose. Once the containment sprays have been successful in sweeping the iodine to the suppression pool, the iodine must be retained in the water. To achieve this, the pH level of the suppression pool will now be raised to 7 or above following the accident, and then maintained at 7 or above. This prevents significant fractions of the dissolved iodine from being converted to elemental iodine and then re-evolving to the containment atmosphere. During the course of the accident the pH of the suppression pool can decrease due to radiolysis of reactor coolant and chloride-bearing electrical insulation, which would create acids. The method for pH control will use the existing Standby Liquid Control (SLC) system for raising (and maintaining) long-term post-accident pH levels to 7 or above. Calculations have shown that the contents of one tank of the Standby Liquid Control solution will be effective in raising and maintaining pH levels for 30 days following the DBA.

Post-accident operator actions are minimized. The operator action associated with initiating the Containment Spray system does not change. Containment Spray is initiated via a push button in the Control Room. The previously required manual initiation of the MSIV–LCS involved multiple operator actions to open and close numerous valves and start the blowers, which will no longer be required. Replacing these actions, the new analysis simply assumes the operator closes the Main Steam Shutoff Valves (which was previously one of the steps in manually initiating the MSIV–LCS system), and based on post-accident pH samples of the suppression pool, initiates the Standby Liquid Control system, which is accomplished via two key lock switches in the Control Room. These operator actions are less complex than those previously required, and minimize the probability of an error.

Other accidents, as described in USAR 15, were reviewed. The original methodology, input parameters and overall conclusions contained within these accident evaluations were found to be unaffected by the changes proposed by this activity. Removing the Technical Specification requirements of the MSIV–LCS and increasing MSIV allowable leakage rates has been addressed in the LOCA reanalysis and does not adversely affect operation of other equipment or systems important to safety. This activity does not alter or impact plant systems, structures or components which were not appropriately addressed in the LOCA reanalysis. No new accident initiator or failure mode is introduced. The physical isolation of the MSIV–LCS from the Main Steam system will eliminate leakage pathways. This modification will be performed as part of the PNPP design change process.

With respect to the change in main steam line leakage limits, the BWROG has concluded, based on an in-depth evaluation of MSIV leakage (as discussed in NEDC–31858 ‘‘BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems,’’ Revision 2, and summarized in NUREG-1169 ‘‘Technical Findings Related to Generic Issue C–8; Boiling Water Reactor Main Steam Isolation Valve and Leakage Treatment Methods’’), that leakage rates of up to 500 scfh are not indicative of substantial mechanical defects in the valves which would challenge the capability of the valves to fulfill their safety function of isolating the steam lines. Therefore, as demonstrated in the design basis LOCA radiological reanalysis, the proposed increased allowable MSIV leakage rate (i.e., each line less than or equal to 100 scfh and total leakage less than or equal to 250 scfh when tested at Pa) will not affect each MSIV's isolation function capability. Additionally, no new operator actions or errors are introduced as a result of the increased main steam line leakage limits, other than those addressed above.

Based on the above discussions, the proposed change would not create the possibility of a new or different kind of accident from any previously evaluated.

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

The worst case LOCA (i.e., a main steam line break upstream of the inboard MSIV) has been reanalyzed using the revised DBA source term (NUREG–1465 and the NEI generic framework document) in order to assess the radiological consequences of the increased MSIV leak rate, and not taking credit for MSIV–LCS. The radiological analyses used conservative assumptions and analytical techniques. The results of the revised DBA source term dose calculations should be determined acceptable using the current licensing basis acceptance limits (those that were used for initial plant licensing).

As noted in the NEI Generic Framework Document (‘‘Generic Framework for Application of Revised Accident Source Term to Operating Plants,’’ EPRI TR–105909, Interim Report, November 1995), ‘‘to demonstrate that an adequate margin of safety is maintained, the licensee may show that the doses associated with the revised design basis (resulting from the revised source term together with the plant change under evaluation) are less than the licensing acceptance limits for the plant.’’

The licensing acceptance limits for off-site dose are discussed in Supplement 8 to the NRC Safety Evaluation Report (SER) for PNPP, Section 15.3, ‘‘Radiological Consequences of Design Basis Accidents.’’ The licensing acceptance limits are the guideline values of 10 CFR 100.11, ‘‘Reactor Site Criteria.’’ The SER states ‘‘The doses computed for this accident are less than the guideline values of 10 CFR 100.11 and the staff concludes that the Perry plant is adequately designed to mitigate the off-site consequences arising from a LOCA.’’ For Control Room doses, the licensing acceptance limit is discussed in Supplement 10 to the NRC SER, Section 6.4, ‘‘Control Room Habitability.’’ The licensing acceptance limits are as stated therein, i.e., ‘‘The staff's LOCA analysis indicates that the Control Room doses are within the guidelines of General Design Criterion (GDC) 19 of Appendix A to 10 CFR part 50 and of Section 6.4 of the Standard Review Plan (SRP, NUREG–0800).’’

The revised PNPP design basis calculations (i.e., the revised DBA source term coupled with the plant changes under evaluation) demonstrated that the resulting off-site and Control Room doses were below the licensing acceptance limits contained in 10 CFR part 100, 10 CFR part 50, Appendix A, General Design Criterion 19, and SRP Section 6.4. An acceptable margin of safety is inherent in these licensing acceptance limits. The improvement in the technical knowledge base and in the analytical techniques that are part of the revised accident source term, and the modeling of the increased MSIV leakages without taking credit for MSIV–LCS, do not alter the acceptability of the margin. Therefore, the resulting calculated Control Room and off-site doses, which are well within regulatory limits, ensure that the proposed change does not involve a significant reduction in the margin of safety.

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Acting Project Director: Ronald R. Bellamy.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440, Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of amendment request: September 3, 1998.

Description of amendment request: The proposed license amendment increases the present Division 3 Diesel Generator (High Pressure Core Spray System) fuel level requirements to account for (1) a rounding error in the calculation, and (2) the unusable volume due to vortex formation at the eductor nozzles located in the fuel oil storage tank.

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 revises the Division 3 Diesel Generator (DG) 7-day fuel oil supply requirement and the 6-day fuel oil supply requirement due to a rounding error in the calculation and due to the consideration of vortex formation near the eductor suction nozzle located near the bottom of the fuel oil storage tank. The proposed change ensures a sufficient DG fuel oil volume to maintain submergence of the eductor suction nozzle so that a vortex formation does not occur. Eliminating the concerns of a vortex formation will provide assurance that the DG fuel oil system will perform its intended function. Analyzed events are initiated by the failure of plant structures, systems, or components. The DGs are not considered as initiators of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase its failure probability of any plant equipment that initiates an analyzed event. As such, the probability of occurrence for a previously analyzed accident is not significantly increased.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed change ensures a sufficient DG fuel oil volume to maintain submergence of the eductor suction nozzle so that a vortex formation does not occur. The proposed change continues to ensure that the DG fuel oil system will adequately support the design basis performance and mitigative function of the DG. The proposed change does not affect the performance of any credited equipment. As a result, no analyses assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

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

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

The proposed change revises the Division 3 DG 7-day fuel oil supply requirement and the 6-day fuel oil supply requirement due to a rounding error in the calculation and due to the consideration of vortex formation near the eductor suction nozzle located near the bottom of the fuel oil storage tank. The proposed change ensures a sufficient DG fuel oil volume to maintain submergence of the eductor suction nozzle so that a vortex formation does not occur. Eliminating the concerns of a vortex formation will provide assurance that the DG fuel oil system will perform its intended function. The proposed change does not involve a physical change to the DG fuel oil system or tank, nor does it change the operating characteristics or the safety function of the DG. The proposed change does not involve a physical alteration of the plant. No new or different equipment is being installed and no installed equipment, which might initiate a new or different kind of accident, is being operated in a different manner. The proposed change does not impact core reactivity or the manipulation of fuel bundles. The DG performs a mitigative function. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

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

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change revises the Division 3 DG 7-day fuel oil supply requirement and the 6-day fuel oil supply requirement due to rounding error in the calculation and due to the consideration of vortex formation near the eductor suction nozzle located near the bottom of the fuel oil storage tank. The margin of safety is being maintained by the proposed change from the margin of safety established by the original design. The proposed change ensures a sufficient DG fuel oil volume to maintain submergence of the eductor suction nozzle so that vortex formation does not occur. Eliminating the concerns of a vortex formation will provide assurance that the DG fuel oil system will perform its intended function. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change, in fact, provides assurance of the DG's ability to perform its intended function as previously evaluated. The proposed change does not significantly impact any safety analysis assumptions or results.

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

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

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

NRC Acting Project Director: Ronald R. Bellamy.

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: September 8, 1998.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 5.3.1, ''Design Features—Reactor Core—Fuel Assemblies.'' A different type of fuel rod cladding would be added. The associated bases would also be changed.

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 has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Further, there are no evaluated accidents in which the fuel cladding or fuel assembly structural components are assumed to arbitrarily fail as an accident initiator. The fuel handling accident assumes that the cladding does, in fact, fail as a result of an undefined fuel handling event. However, the probability of that undefined initiating event is independent of the properties of the fuel rod cladding.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Therefore, in both non-LOCA and LOCA accident scenarios, there will be no significant increase in cladding failure or fission product release.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. Therefore, M5 fuel cladding and fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different kind of accident.

3. Not involve a significant reduction in a margin of safety because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. The M5 alloy is expected to perform similarly to Zircaloy-4 for all normal operating and accident scenarios, including both non-LOCA and LOCA scenarios. For LOCA scenarios, where the slight differences in M5 material properties relative to Zircaloy-4 could have

some impact on the overall accident scenario, plant-specific LOCA analyses will be performed prior to the use of batch quantities of fuel assemblies containing either fuel rod cladding, fuel rod end plugs, or fuel assembly structural components fabricated from M5. These plant-specific LOCA analyses, required by TS 6.9.1.7, ''Core Operating Limit Report,'' will either demonstrate that all current, applicable, and appropriate margins of safety will be maintained during the use of the M5 alloy or their results will be submitted for NRC review and approval prior to use of the M5 alloy.

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.

Local Public Document Room location: University of Toledo, William Carlson Library, Government Documents Collection, 2801 West Bancroft Avenue, Toledo, OH 43606.

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

NRC Acting Project Director: Ronald R. Bellamy.

Yankee Atomic Electric Company, Docket No. 50–029, Yankee Nuclear Power Station, Franklin County, Massachusetts

Date of amendment request: August 20, 1998.

Description of amendment request: By letter dated August 20, 1998, the licensee submitted a License Amendment request related to three Technical Specification (TS) administrative changes. The first is to remove a definition from the DEFINITIONS section of the TS that is provided in 10 CFR part 20. The second change is to transfer the site map from Section 5.0 of the TS to the Final Safety Analysis Report and to replace the map with a textual description of the site location. Lastly, to delete TS 5.1.1—EXCLUSION AREA.

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

The proposed changes are administrative in nature and in no way affect the safety of the Yankee Nuclear Power Station (YNPS). The proposed deletion of the definition for SITE BOUNDARY in no way reduces or eliminates any regulatory requirement which Yankee Atomic Electric Company must currently satisfy. Likewise, the relocation of the YNPS site map from the YNPS Technical Specifications to the YNPS Final Safety Analysis Report is devoid of any safety implications. Therefore, the proposed changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The administrative nature of the changes will not affect safety related systems or components and, therefore, involve no significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different accident from any previously evaluated. The proposed changes do not modify any plant systems or components and, therefore, do not create the possibility of a new or different accident from any previously evaluated.

3. Involve a significant reduction in the margin of safety. The proposed changes do not involve any physical changes to the plant nor any changes in plant procedures. Therefore, there will be no 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.

Local Public Document Room location: Greenfield Community College, 1 College Drive, Greenfield, Massachusetts 01301.

Attorney for licensee: Thomas Dignan, Esquire, Ropes and Gray, One International Place, Boston, Massachusetts 02110–2624.

NRC Project Director: Seymour H. Weiss.

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.

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

Date of amendment request: August 27, 1998.

Brief description of amendment request: The amendment revises Technical Specifications 3.0.4 and 4.0.4 to be consistent with the guidance provided in Generic Letter 87–09 dated June 4, 1987.

Date of publication of individual notice in **Federal Register**: September 8, 1998 (63 FR 47529).

Expiration date of individual notice: October 8, 1998.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of amendment request: August 21, 1998.

Description of amendment request: The amendment would remove the requirement for the Automatic Depressurization System function of the Electromatic Relief Valves to be operable during Reactor Vessel Pressure Testing. Additionally, note h of Table 3.1.1 will be corrected due to a typographical error introduced in the issuance of Amendment 75.

Date of publication of individual notice in **Federal Register**: September 10, 1998 (63 FR 48527).

Expiration date of individual notice: October 13, 1998.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Wisconsin Public Service Corporation, Wisconsin Power and Light Company and Madison Gas and Electric Company, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, WI

Date of application for amendment: April 8, 1998, modified by letter dated August 27, 1998.

Brief description of amendment request: The proposed amendment would reduce the maximum allowable level of reactor coolant system activity (dose equivalent 1–131) to provide a means of accepting higher projected leak rates for steam generator tubes while still meeting offsite and control room dose criteria. Also included is a change to the secondary coolant activity level for which an increased sampling frequency applies.

Date of publication of individual notice in **Federal Register**: September 14, 1998 (63 FR 49137).

Expiration date of individual notice: October 14, 1998.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001.

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 at the local public document rooms for the particular facilities involved.

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

Date of application for amendment: March 12, 1998, as supplemented August 14, 1998. The August 14, 1998, supplemental letter provided clarifying information only, and did not change the initial no significant hazards consideration determination.

Brief description of amendment: This amendment deletes Technical Specification surveillance requirement 4.9.12.d.4, which requires verification at least once every 18 months that the Fuel Handling Building Emergency Exhaust System filter cooling bypass valve is locked in the balanced position.

Date of issuance: September 11, 1998.

Effective date: September 11, 1998.

Amendment No: 82.

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

Date of initial notice in **Federal Register**: April 8, 1998 (63 FR 17222).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

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

Date of application for amendment: January 28, 1998 (NRC–98–0003) as supplemented March 10, 1998

Brief description of amendment: The amendment revises technical specification (TS) 3.4.10, TS Figure 3.4.10–1, and the associated bases by changing the prohibited and restricted operating region associated with core thermal-hydraulic stability. Also, TS 3.4.1.4, TS Figure 3.4.1.4–1, and the associated bases are revised to reflect stability-related improvements in operating restrictions for idle recirculation loop startup. Finally, in an unrelated change, TS Tables 3.3.7.5–1 and 4.3.7.5–1 are revised to delete neutron flux from the list of accident monitoring instrumentation in TS 3.3.7.5.

Date of issuance: September 16, 1998

Effective date: September 16, 1998, with full implementation within 90 days.

Amendment No.: 128.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: February 25, 1998 (63 FR 9598). The March 10, 1998, letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of application for amendment: June 5, 1998 (NRC–98–0067), as supplemented August 24, 1998.

Brief description of amendment: The amendment revises Technical Specification (TS) 2.1.2, ''Thermal Power, High Pressure and High Flow,'' by changing the values for the safety limit minimum critical power ratio from 1.09 to 1.11 for two recirculation loop operation and from 1.11 to 1.13 for single recirculation loop operation for Cycle 7. The amendment also revises the footnote to TS 2.1.2 to indicate that these revised values are applicable for Cycle 7 operation only.

Date of issuance: September 21, 1998.

Effective date: September 21, 1998, with full implementation prior to restart from the sixth refueling outage.

Amendment No.: 129.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35988). The August 24, 1998, letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 21, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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

Date of application for amendments: May 8, 1998.

Brief description of amendments: The amendments revise the Power Range Neutron Flux Trip setponts in the event of inoperable main steam safety valves.

Also, the amendments delete the reference to three-loop operation. These changes are consistent with the proposed Improved Standard Technical Specifications submitted by the licensee on May 27, 1997.

Date of issuance: September 17, 1998.

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

Amendment Nos.: Unit 1—181; Unit 2—163.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 29, 1998 (63 FR 40554).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 17, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

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

Date of application for amendments: October 6, 1998, as supplemented by letter dated August 24, 1998.

Brief description of amendments: The amendments delete all references to the steamline low pressure safety injection function.

Date of issuance: September 22, 1998.

Effective date: As of the date of issuance to be implemented in the refueling outage associated with the plants' hardware modifications.

Amendment Nos.: Unit 1—182; Unit 2—164.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: November 19, 1997 (62 FR 61841).

The August 24, 1998, submittal provided clarifying information that did not change the scope of the original **Federal Register** notice, and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: J. Murrey Atkins Library, University of North Carolina at Charlotte, 9201 University City Boulevard, Charlotte, North Carolina.

Entergy Operations, Inc., Docket Nos. 50–313 and 50–368, Arkansas Nuclear One, Units 1 and 2, Pope County, Arkansas

Date of amendment request: October 2, 1996, as supplemented by the letter dated June 18, 1997.

Brief description of amendments: The amendments relocate the Radiological Effluents Technical Specifications (RETS) to the Offsite Dose Calculation Manual and the Process Control Program. The NRC provided guidance to all power reactors licensees and applicants on the proposed TS changes in Generic Letter 89–01, ''Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program,'' dated January 31, 1989.

Date of issuance: September 23, 1998.

Effective date: September 23, 1998.

Amendment Nos.: Unit 1; 193 and Unit 2; 193.

Facility Operating License Nos. DPR–51 and NPF–6: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: January 15, 1997 (62 FR 2188).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

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

Date of application for amendments: March 3, 1998.

Brief description of amendments: The amendments modify surveillance requirement 4.6.4.2.b.4 for Unit 1 and the Technical Specification bases 3/4.6.4 for Unit 1 and 2.

Date of issuance: September 17, 1998.

Effective date: September 19, 1998, with full implementation within 45 days.

Amendment Nos.: 223 and 207.

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

Date of initial notice in **Federal Register**: July 1, 1998 (63 FR 35990).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 17, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

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

Date of application for amendment: July 16, 1998, as supplemented September 3, 1998. The application dated July 16, 1998, supersedes a July 2, 1997, application in its entirety.

Brief description of amendment: The amendment changes Technical Specification 3/4.2.3 regarding reactor coolant chemistry in accordance with a report by Electrical Power Research Institute, Inc., TR–103515–R1, ''BWR Water Chemistry Guidelines, 1996 Revision.''

Date of issuance: September 18, 1998.

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

Amendment No.: 163.

Facility Operating License Nos. DPR–63 and NPF–69: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register**: August 13, 1998 (63 FR 43432).

The September 3, 1998, submittal contained clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 18, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Project Director: S. Singh Bajwa.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: August 23, 1995.

Brief description of amendment: The amendment extends the Technical Specification (TS) Allowed Outage Time (AOT) for an inoperable Safety Injection Tank (SIT) from 1 hour to 24 hours, unless the SIT is inoperable due to either boron concentration not within its limits or an inoperable water level or nitrogen cover pressure instrument. The proposed change, for these two special cases, extends the AOT for an inoperable SIT to 72 hours. In addition, the completion times and conditions for action statements and the criteria for surveillance requirements are changed. The TS Bases are also updated to reflect the changes.

Date of issuance: September 3, 1998.

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

Amendment No.: 220 Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: September 13, 1995 (60 FR 47621).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 3, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

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

Date of application for amendment: August 6, 1998, as supplemented September 3 and 21, 1998.

Brief description of amendment: The latest Millstone Unit 3 steam generator tube inspection began on September 24, 1996, and was completed on October 1, 1996. The inspection results placed the steam generators in Category C–2. Technical Specification Surveillance Requirement 4.4.5.3.a establishes an allowable inspection interval of 24 calendar months for this category. Without an extension of the interval, Millstone Unit 3 must shut down prior to September 24, 1998, to perform the necessary inspections. The amendment allows a one-time extension to the surveillance interval until the next refueling outage or July 1, 1999, whichever date is earlier.

Date of issuance: September 23, 1998.

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

Amendment No.: 163.

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

Date of initial notice in **Federal Register**: August 17, 1998 (63 FR 43964).

The September 3 and 21, 1998, letters provided clarifying information that did not change the scope of the August 6, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 23, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: July 26, 1996, as supplemented September 5, 1997, as revised December 4, 1997, and as supplemented March 6, March 26, April 8, April 17, April 22, May 5, May 12, May 29, June 15, July 1, July 20, and July 30, 1998.

Brief description of amendment: The amendment revises the operating license and the Technical Specifications to allow increase of the maximum reactor core thermal power level from 1670 megawatts-thermal (MWt) to 1775 MWt.

Date of issuance: September 16, 1998.

Effective date: September 16, 1998. Full implementation within 90 days of issuance.

Amendment No.: 102.

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

Date of publication of individual notice in **Federal Register**: February 25, 1998 (63 FR 9606).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 16, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

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

Date of application for amendments: February 27, 1998, as supplemented July 14, 1998.

Brief description of amendments: The amendments allow a design modification of the existing Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC). The design modification installs a Diverse Scram System (DSS) designed to meet the requirements of a DSS described by 10 CFR 50.62 (ATWS Rule) for non-Westinghouse designed plants and make major modifications to the existing AMSAC.

Date of issuance: September 22, 1998.

Effective date: September 22, 1998, with full implementation by the completion of the next scheduled refueling outage.

Amendment Nos.: 138 and 129.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the license to authorize a design modification of the existing Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC).

Date of initial notice in **Federal Register**: August 17, 1998 (63 FR 43965).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 22, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Philadelphia Electric Company, Docket No. 50–171, Peach Bottom Atomic Power Station, Unit 1, York County, Pennsylvania

Date of amendment request: March 2, 1998.

Brief description of amendment: This amendment will revise the Peach Bottom Atomic Power Station, Unit 1, Technical Specifications (TS) to include requirements for control of effluents and annual reporting in accordance with the requirements of 10 CFR 50.36a.

Date of issuance: September 14, 1998.

Effective date: As of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

Amendment No.: 9.

Facility Operating License No. DPR–12: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register**: July 1, 1998 (61 FR 35994). The NRC's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

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

Date of amendment request: August 28, 1998.

Brief description of amendments: The amendments modified TS 4.0.5 to state that the inservice testing requirement for exercise testing in the closed direction for specified Unit 1 containment isolation valves shall not be required until the next plant shutdown to Mode 5 of sufficient duration to allow the testing or until the next refueling outage scheduled in March 1999.

Date of issuance: September 24, 1998.

Effective date: September 24, 1998, to be implemented within 7 days.

Amendment Nos.: Unit 1—Amendment No. 95; Unit 2—Amendment No. 82.

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

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes (63 FR 48254). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by October 8, 1998, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendments.

The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of NSHC are contained in a Safety Evaluation dated September 24, 1998.

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

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

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

Date of amendment request: February 16, 1998, as supplemented April 2, July 15, and August 13, 1998.

Brief description of amendments: The amendments revised TS 3/4.4.5, ''Steam Generators,'' and its Bases to allow the implementation of 1-volt voltage-based repair criteria for the steam generator tube support plate-to-tube intersections for Unit 2 in accordance with Generic Letter 95–05, and made related Unit 1 administrative changes for consistency of wording (the Nuclear Regulatory Commission (NRC) had previously approved a similar 1-volt voltage-based repair criteria application for Unit 1). In addition, the amendments made an administrative change to Bases 3/4.4.6.2, ''Operational Leakage,'' to clarify that the allowable steam generator leakage specification applies to both Unit 1 and Unit 2.

Date of issuance: September 24, 1998.

Effective date: September 24, 1998, to be implemented within 30 days.

Amendment Nos.: Unit 1—Amendment No. 96; Unit 2—Amendment No. 83.

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

Date of initial notice in **Federal Register**: May 20, 1998 (63 FR 27765).

The additional information contained in the supplemental letters dated July 15 and August 13, 1998, were clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 24, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50–440 Perry Nuclear Power Plant, Unit 1, Lake County, Ohio

Date of application for amendment: August 29, 1995, supplemented June 25, 1998

Brief description of amendment: This amendment revises Technical Specification Tables 3.3.5.1–1, ''Emergency Core Cooling System Instrumentation,'' and 3.3.6.1–1, ''Primary Containment and Drywell Isolation Instrumentation,'' by revising allowable values for selected plant process instrumentation in accordance with Instrument Setpoint Methodology Group and GE Topical Report NEDC–31336, ''General Electric Instrument Setpoint Methodology,'' dated October 1986.

Date of issuance: September 15, 1998.

Effective date: September 15, 1998.

Amendment No.: 93.

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

Date of initial notice in **Federal Register**: December 6, 1995 (60 FR 62496)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 1998.

No significant hazards consideration comments received: No.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, OH 44081.

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

Date of application for amendment: June 1, 1998, supplemented July 14, 1998

Brief description of amendment: The changes revise the F* and elevated F* (EF*) criteria used to disposition indications in the roll expansion joint of degraded steam generator (SG) tubes within the tubesheet.

Date of issuance: September 22, 1998.

Effective date: September 22, 1998.

Amendment No.: 138.

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

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 22, 1998. .

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, WI 54311–7001

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

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

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

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

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

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

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

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

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 6, 1998, 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 at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

Entergy Operations Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: September 18, 1998, as superseded by letter dated September 23, 1998.

Description of amendment request: The amendment changes the Appendix A TSs by revising Note ''1'' in Table 2.2–1, ''Reactor Protective Instrumentation Trip Setpoint Limits'' and Note ''a'' in Table 3.3–1, ''Reactor Protective Instrumentation,'' both applicable to high logarithmic power reactor trip instrumentation. Additionally, the requested changes clarify the terms RATED THERMAL POWER and THERMAL POWER used in Tables 2.2–1, 3.3–1 and 4.3–1. A Bases change is made to support these changes.

Date of issuance: September 24, 1998.

Effective date: September 24, 1998.

Amendment No: 145.

Facility Operating License No. NPF–38: Amendment revises the Technical Specifications Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 24, 1998.

Local Public Document Room Location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street NW., Washington, D.C. 20005–3502.

NRC Project Director: John N. Hannon.

Southern California Edison Company, et al., Docket No. 50–361, San Onofre Nuclear Generating Station, Unit No. 2, San Diego County, California

Date of application for amendment: September 22, 1998.

Brief description of amendment: The amendment revises the technical specifications (TS) to change the operative parameter for setting and removing the operating bypass bistables for Logarithmic Power Level—High, Reactor Coolant Flow—Low, Local Power Density—High, and Departure from Nucleate Boiling Ratio—Low trips. The operative parameter specified in the TS is being changed from ''THERMAL POWER'' to logarithmic power.

Date of issuance: September 25, 1998.

Effective date: September 25, 1998.

Amendment No.: 142.

Facility Operating License No. NPF–10: The amendment revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No. The Commission's related evaluation of the amendments, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 25, 1998.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, P.O. Box 800, Rosemead, California 91770.

Local Public Document Room location: Main Library, University of California, Irvine, California 92713

Dated at Rockville, Maryland, this 30th day of September 1998.

For the Nuclear Regulatory Commission.

John N. Hannon,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 98–26746 Filed 10–6–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Notice of Issuance of Revised NRC Form 3; Notice to Employees

The Nuclear Regulatory Commission has issued a revised NRC Form 3, ''Notice to Employees'', dated September 1998, effective October 7, 1998. The Form has been revised to reflect the closure of the NRC field office located in Walnut Creek, California, effective close of business, September 30, 1998. Individuals who have been reporting concerns to the Walnut Creek field office should now report their concerns to the NRC's Region IV office located in Arlington, Texas. The toll-free number for the Arlington, Texas office is (800) 952–9677.

A copy of NRC Form 3 has been placed in the NRC's Public Document Room, the Gelman Building, 2120 L Street, NW. (Lower Level), Washington, DC 20037, for review and copying by interested persons.

Dated at Rockville, Maryland, this 1st day of October 1998.

For the Nuclear Regulatory Commission.

Edward T. Baker, III,

Agency Allegation Advisor, Office of the Director, Office of Nuclear Reactor Regulation.

[FR Doc. 98–26851 Filed 10–6–98; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Extension; Comment Request

Upon Written Request, Copy Available From: Securities and Exchange Commission, Office of Filings and Information Services, 450 Fifth Street, NW, Washington, D.C. 20549

Extension:

Form S–6—File No. 270–181—OMB Control No. 3235–0184

Notice is hereby given that, pursuant to the Paperwork Reduction Act of 1995 [44 U.S.C. 3501 *et seq.*], the Securities and Exchange Commission