NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 31, 1999, through January 14, 2000. The last biweekly notice was published on January 12, 2000.

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

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

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

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

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

By February 25, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: December 17, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification 2.1.1.2 to incorporate cycle-specific safety limit minimum critical power ratios (SLMCPRs) for the core that will be loaded during the upcoming refueling outage

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 license amendment establishes a revised SLMCPR value of 1.07 for two recirculation loop operation and 1.09 for single recirculation loop operation. The derivation of the cycle-specific SLMCPRs was performed using NRC approved methods and uncertainties described in Amendment Number 25 to NEDE–24011–P–A (GESTAR II) and Licensing Topical Reports NEDC–32601P–A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations" and NEDC–32694P–A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation."

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established, consistent with NRC approved methods, to ensure that fuel performance during normal, transient, and accident conditions is acceptable.

The probability of an evaluated accident is not increased by revising the SLMCPR values. The change does not require any physical plant modifications or physically affect any plant components. Therefore, no individual precursors of an accident are affected.

The proposed license amendment establishes a revised SLMCPR that ensures that the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences. Specifically, the reload analysis demonstrates that a SLMCPR value of 1.07 (1.09 for single loop operation) ensures that less than 0.1 percent of the fuel rods will experience boiling transition during any plant operation if the limit is not violated.

Based on (1) the determination of the new SLMCPR values using NRC approved methods and uncertainties, and (2) the operability of plant systems designed to mitigate the consequences of accidents not having been changed; the consequences of an accident previously evaluated have not been increased.

Therefore, the proposed Technical Specification change does not involve an 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 license amendment involves a revision of the SLMCPR from 1.11 to 1.07 for two recirculation loop operation and from 1.13 to 1.09 for single loop operation based on the results of analysis of the Cycle 8 core which will once again be fully loaded with GE11 fuel. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in the allowable methods of operating the facility. This proposed license amendment does not involve any modifications of the plant configuration or changes in the allowable methods of operation. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed license amendment establishes a revised SLMCPR value of 1.07 for two recirculation loop operation and 1.09 for single recirculation loop operation. The derivation of these revised SLMCPRs was performed using NRC approved methods and uncertainties described in Amendment Number 25 to NEDE–24011–P–A (GESTAR II) and Licensing Topical Reports NEDC–32601P–A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations" and NEDC–32694P–A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation." Use of these methods ensures that the resulting SLMCPR satisfies the fuel design safety criteria that less than 0.1 percent of the fuel rods experience boiling transition if the safety limit is not violated. Based on the assurance that the fuel design safety criteria will be met, the proposed license amendment does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 17, 1999.

Description of amendment request: The proposed amendment would revise Technical Specification Surveillance Requirement (SR) 3.6.1.3.9 to relax the SR frequency by allowing a representative sample of excess flow check valves (EFCVs) to be tested every 18 months, such that each EFCV will be tested at least once every 10 years. Current SR 3.6.1.3.9 requires all EFCVs to be tested every 18 months.

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

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

The current SR frequency requires each reactor instrumentation line EFCV to be tested every 18 months. The EFCVs at Fermi 2 are designed to close automatically in the event of a line break downstream of the valve. Indicating lights on a control room panel monitor EFCV positions. These valves may be reopened by actuation of a solenoid valve, which is operated from a local control panel. EFCVs at Fermi 2 are designed and installed following the guidance of Regulatory Guide 1.11. This proposed change allows a reduced number of EFCVs to be tested every 18 months. Industry operating experience, documented in BWROG [Boiling Water Reactor Owners Group] Report B21–00658–01, concludes that a change in surveillance test frequency has a minimal impact on the reliability for these valves. A failure of an EFCV to isolate cannot initiate previously evaluated accidents; therefore, there can be no increase in the probability of occurrence of an accident as a result of this proposed change.

Fermi 2 UFSAR [Updated Final Safety Analysis Report], Subsection 15.6.2 evaluates an instrument line pipe break within secondary containment. The evaluation assumes that a small instrument line instantaneously and circumferentially breaks at a location where it may not be possible to isolate it and where immediate detection is not automatic or apparent. The evaluation concluded that pressurization of the secondary containment would not result from an instrument line break and a failure of the associated EFCV to isolate the ruptured line. The standby gas treatment system is not impaired by this event, and the calculated offsite exposure is substantially below the guidelines of 10 CFR 100. Additionally, coolant lost from such a break is inconsequential when compared to the makeup capabilities of the feedwater or RCIC [reactor core isolation cooling] system. The BWROG report concludes that the risk to the public with the extended testing interval is several orders of magnitudes below the general public annual exposure limits in 10 CFR 20.105.

Although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by the analysis in the UFSAR. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the radiological consequences of an accident previously evaluated as a result of this change.

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

This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in the BWROG report provides supporting evidence that the reduced testing frequency will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduction in test frequency is bounded by the evaluation of an instrument line pipe break described in Subsection 15.6.2 of the UFSAR. This change is not a physical alteration of the plant and will not alter the operation of the structures, systems and components as described in the UFSAR. Therefore, a new or different kind of accident will not be created.

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

The consequences of a postulated instrument line pipe break have been evaluated in Subsection 15.6.2 of the UFSAR. The evaluation assumed the line instantaneously and circumferentially breaks at a location where it may not be possible to isolate it and that the EFCV fails to isolate the break. Therefore, any potential failure of an EFCV as a result of the reduced testing frequency is bounded by this evaluation and does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Section Chief: Claudia M. Craig

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: August 18, 1999.

Description of amendment request: The proposed change would amend Technical Specification 3.5.3 and its associated Bases to reflect a change in the reactor coolant system (RCS) low pressure setpoint for Arkansas Nuclear One, Unit 1 (ANO–1). The RCS low pressure setpoint has been adjusted in the conservative direction to account for both the uncertainties associated with the actual value and the current number of plugged steam generator tubes.

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:

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

The proposed change to raise the current technical specification (TS) ESAS [Engineered Safeguards Actuation Signal] setpoint for low RCS pressure does not require new hardware or physical equipment modifications to the plant design. By raising the setpoint, a more prompt actuation of associated safeguards equipment will be achieved for the accident scenarios previously analyzed in the ANO–1 Safety Analyses Report (SAR). A more expeditious actuation will ensure a more timely response to the accident and serve to potentially decrease the consequences of an accident. The RCS Pressure LO LO [Low Low] alarm setpoint has also been raised and applicable procedures revised to provide the operator sufficient time to bypass the actuation during controlled plant maneuvers.

Therefore, the raising of the low RCS pressure ESAS setpoint from 1526 psig [pounds per square inch, guage] to 1585 psig does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change is relevant to accident response and mitigation and has no [a]ffect on accident initiation. An inadvertent actuation of the HPI [high pressure injection] system could result in pressurizing the RCS to the point where a pressurizer safety valve could open and subsequently fail to close, resulting in a loss of coolant accident.

However, this event remains unaffected for normal power operations and requires discussion of depressurization events only, such as a planned cooldown, when an inadvertent actuation could occur earlier due to the proposed higher setpoint. This concern is mitigated by the increase of the RCS Pressure LO LO alarm setpoint from approximately 1550 psig to 1640 psig, thus providing the operator ample time to bypass the low RCS pressure ESAS setpoint prior to inadvertent actuation. Therefore, no new, previously unevaluated event has been introduced relating to the inadvertent actuation of HPI components due to the proposed change.

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

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

The proposed change conservatively raises the existing low RCS pressure ESAS setpoint to a new value using existing installed equipment. The new value provides protection for the entire spectrum of break sizes based on applicable evaluations and considers the effects of projected steam generator tube plugging activities. The setpoint is also sufficiently below normal operating pressure to aid in preventing spurious initiation.

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

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations, Inc. has determined that the requested change does *not* involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 16, 1999.

Description of amendment request: The proposed changes to the Arkansas Nuclear One, Units 1 and 2 (ANO–1 and ANO–2), Technical Specifications (TSs) and associated Bases would provide a 30-day allowable outage time (AOT) for Startup Transformer No. 2 (SU#2) which is an offsite power source shared by both units. This 30-day AOT would be used infrequently for the purpose of performing preventative maintenance on the transformer to increase its reliability. The current TS constraints would require both units to be in cold shutdown in order to perform this maintenance. In addition, changes have been requested to the requirements associated with demonstrating the operability of the emergency diesel generators to increase the reliability of this power supply.

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:

An evaluation of the proposed changes has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

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

Based on existing methodologies, guidance, and procedures utilized at ANO, including required assessments of risk associated with any significant maintenance activity, the provision of a 30-day AOT for preplanned preventative maintenance on SU#2 is acceptable. The resulting increase in overall risk was considered to fall into NRC Risk Region III (''Very Small Change''). Additionally, removal of SU#2 from service in any plant mode of operation has been previously evaluated and found acceptable given the existing guidance and regulations associated with offsite power sources.

Five offsite power feeds are available to the ANO switchyard with no more than two of the feeds in close proximity to one another for a given length, except within the switchyard itself. Failure of one feed, regardless of the cause, will result in no more than one additional failure, leaving at least three offsite power sources yet available, assuming the failure remains outside the ANO switchyard. For events that pose a threat within the ANO switchyard, four redundant Class 1E EDGs [emergency diesel generators] and one Alternate AC [alternating current] diesel generator are capable of supply power to the units. Upon loss of the remaining offsite power transformer of a unit which may be off-line, offsite power may be restored via backfeed operations from the Main Transformers to the Unit Auxiliary transformer to supply in-house loads. This ensures the availability of redundant power sources, including the applicable contingencies established during safety-related equipment maintenance performed at ANO, are sufficient in maintaining safe unit operations during preplanned preventative maintenance on SU#2 transformer. Therefore, providing a 30-day AOT for preplanned preventative maintenance on SU#2, not to be applied more than once in any 10-year period, does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

The elimination of excessive EDG operability demonstrations (cold starts) during periods when another required power source is inoperable acts to enhance overall EDG reliability and is consistent with guidance provided in NRC Generic Letter 84–15 ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability'' and the Revised Standard Technical Specifications (NUREG–1430 and –1432). Verification of the operability of the remaining EDG will be performed within 24 hours should the failure mechanism that caused the inoperability of the redundant EDG be concluded to be a common cause type failure. The start test in the latter case acts to ensure that an EDG source remains available when the cause of the failure of the redundant EDG might impact the remaining EDG.

Therefore, eliminating excessive EDG cold starts does *not* involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The removal of SU#2 from service to support needed maintenance activities has been previously evaluated for all modes of plant operation. Extending the current AOT to 30 days on a limited basis does not result in any new accident initiator. The EDGs are not considered accident initiators, but are designed to support mitigation of accident scenarios. The elimination of excessive EDG cold starts acts to enhance overall EDG reliability and has no effect on accident development.

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

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

The associated probabilistic risk assessments indicate that the proposed 30-day AOT for SU#2 does not involve a significant increase in overall site risk, nor reduce the margin to safety. Thorough contingency action planning, which acts to maintain the operability of other equipment important to safety during the SU#2 maintenance window, additionally acts to ensure the margin to safety is maintained. The EDGs are important to safety in that they are designed to supply power to safety system components and equipment during a loss of offsite power. The elimination of excessive cold starts of the EDGs acts to enhance the overall reliability of the EDGs and, therefore, proper mitigation of accident scenarios is likewise enhanced.

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

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations, Inc. has determined that the requested change does *not* involve a significant hazards consideration.

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: December 16, 1999.

Description of amendment request: The proposed license amendment request would revise Fuel Handling Accident (FHA) dose calculations for 3 scenarios documented in the River Bend Station, Unit 1 (RBS), Updated Safety Analysis Report (USAR). The first is a FHA in the fuel building, assumed to occur 24 hours post-shutdown. A second FHA analysis was prepared to support Amendment 35 to RBS Technical Specifications (TS) which assumed a FHA occurs in the primary containment 80 hours post-shutdown during Local Leakage Rate Testing (LLRT). A third analysis was prepared in support of Amendment 85 to the RBS TS which assumed the containment is open at 11 days.

These analyses are being updated to account for several changes. The primary reason for the revisions, as stated by the licensee, was to update the analyses to reflect current RBS operating strategies and make the analyses consistent with each other. Specifically, Cases 1 and 2 of the three analyses assumed a Radial Peaking Factor (RPF) of 1.5 consistent with Regulatory Guide (RG) 1.25. However, current core design strategies could lead to an RPF as high as 1.65. In addition, to account for the potential impact of extended burnup fuel in future operating cycles, an increased iodine-131 gap fraction of 0.12 was more conservatively assumed in lieu of the 0.10 recommended by RG 1.25. The revised analysis also includes a change to the control room atmospheric dispersion factors ($_\chi$/Q) for the Main Control Room (MCR) ventilation system. Credit is taken for Standard Review Plan (SRP) Section 6.4 guidance for manual dual control room air intakes in that the $_\chi$/Q's are divided by 4. The revised FHA analyses also credit this action at a 20 minute delay to be consistent with the Loss of Coolant Accident (LOCA) analysis.

Furthermore, an error was discovered in one of the FHA calculations. The release rate assumed in the analysis did not ensure that the RG 1.25 assumption of a 2-hour release was preserved. The error is the result of an inherent bias in the secondary mixing effects in the dose calculation. The results continue to be bounded by the guidance contained in SRP 15.7.4 and RG 1.25.

Reanalysis showed that the release rate error, compounded with the other changes discussed above, resulted in calculated doses greater than those currently found in the RBS USAR. In addition, some of the doses were also greater than those presented in the Amendment 85 submittal. However, the licensee has stated that the results of the revised analyses remain "well within" 10 CFR 100, the guidance contained in SRP 15.7.4, and RG 1.25. Since the analyses results are above those reported in the RBS USAR, the criterion of 10 CFR 50.59(a)(2)(i) is, therefore, satisfied. Accordingly, the licensee has concluded that these changes involve an unreviewed safety question.

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 significantly increase the probability or consequences of an accident previously evaluated.

The analyses changes described by this proposed change to the USAR are not initiators to events, and, therefore, do not involve the probability of an accident. The changes to the FHA calculations for radiological doses following a FHA reflect the current operating strategies and make the analyses consistent. These changes included:

- accounting for the impact of extended burnup fuel,
- addressing a change to the control room atmospheric dispersion factors assumed in the analysis, and
- revising the Radial Peaking Factor (RPF) used in the analysis. Current core design strategies could lead to a RPF higher [than] that assumed in Regulatory Guide 1.25.

The TRANSACT code is used for offsite dose and control room dose calculations. The TRANSACT code is derived from the TACT V code documented in NUREG/CR–5109. RBS has benchmarked the TRANSACT code as discussed in the request dated August 17, 1995, (RBG–41728) which resulted in the NRC granting Amendment 85.

The revisions to the FHA are used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions include:

- initial fuel movement in the Fuel Building 24 hours after shutdown,
- fuel movement in Primary Containment after 80 hours with leakrate testing being conducted, and
- fuel movement in Primary Containment with the Primary Containment open.

Because the analyses affected by the changes are not considered an initiator to any previously analyzed accident, these changes cannot increase the probability of any previously evaluated accident. Therefore, this change does not increase the probability of occurrence of an accident evaluated previously in the safety analysis report (SAR).

This proposed change to the USAR does increase the consequences of an accident, but the increase is within all regulatory limits and guidance. While the calculated off-site and control room doses of a FHA did increase, the dose consequences remain below the regulatory limits of 10 CFR 100 and 10 CFR 50, Appendix A, GDC [General Design Criterion]–19 as approved per NUREG–0989, and the guidance contained in SRP 15.7.4 of less than 25% of the 10 CFR 100 limits. The cause of these events remains the failure of the fuel assembly lifting mechanism. These analyses demonstrated that for the worst case bundle drop, the regulatory dose guidelines of SRP 15.7.4 continue to be satisfied for the required decay periods.

This change accounts for the potential effects of current fuel design and operating strategies including increased burnup of fuel, increased iodine-131 fraction released, Main Control Room ventilation system operation, and release rate timing assumptions. Reanalysis of the off-site dose calculation demonstrates that the revised doses are increased but remain less than the regulatory limits of 10 CFR 100 and within the guidance of SRP 15.7.4. Therefore, this change does not significantly increase the consequences of an accident previously evaluated in the SAR.

The proposed changes, in conjunction with existing administrative controls, bound the conditions of the current design basis fuel handling accident analysis. The analysis also concludes the limiting offsite radiological consequences are well within the acceptance criteria of NUREG[–]0800, Section 15.7.4 and 10 CFR 50, Appendix A, GDC[–]19. The analysis is also conducted in a conservative manner containing margins in the calculation of mechanical analysis, iodine inventory, and iodine decontamination factor. Each of these conservatisms will further decrease the consequences. Therefore, the proposed changes do not significantly increase the probability or consequences of any previously evaluated accident.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous[ly] analyzed.

This change does not involve initiators to any events in the SAR, nor does the activity create the possibility for any new accidents. Rather, this change is a result of the evaluation of the most limiting FHA, which can occur at River Bend.

The proposed changes to the dose analyses are consistent with previous limits, only revising previous evaluations to account for current operating strategies and assumptions. These changes included:

• accounting for the impact of extended burnup fuel,

• addressing a change to the control room atmospheric dispersion factors assumed in the analysis, and

• revising the Radial Peaking Factor (RPF) used in the analysis. Current core design strategies could lead to a RPF higher [than] that assumed in Regulatory Guide 1.25.

The radiological consequences remain within accepted limits of 10 CFR 100 and guidance of the Standard Review Plan (NUREG–0800) Section 15.7.4. Therefore, these changes are consistent with the design basis analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previous[ly] analyzed.

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

The dose consequences are calculated in accordance with regulatory guidance found in Regulatory Guide 1.25 and the SRP [S]ection 15.7.4. The RBS analyses conservatively assumed that failures are consistent with those in the standard General Electric GESTAR II. These analyses result in a bounding number of fuel failures. The RBS analyses are also consistent with those approved by the NRC [Nuclear Regulatory Commission] in support of Technical Specification Amendments 35 and 85 to the River Bend Station license (NPF–47). The radiological dose consequences resulting from these failures are therefore analyzed using accepted methods and criteria. In addition, the analyses contain known conservatisms and margins to ensure the results will remain bounding.

The revised limits are used to establish operational conditions where specific activities represent situations where significant radioactive releases can be postulated. These operational conditions are consistent with the design basis analysis and are established such that the radiological consequences are at or below the current regulatory limits and guidance. Safety margins and analytical conservatisms have been evaluated and are well understood. Conservative methods of analysis are maintained through the use of accepted methodology and benchmarking the proposed methods to previous analysis. Margins are retained to ensure that the analysis adequately bounds all postulated event scenarios. The proposed change only eliminates some excess conservatism from the analysis.

In addition, EOI [Entergy Operations, Inc.] has implemented NUMARC [Nuclear Management and Resources Council (now NEI)] 91–06 guidelines for shutdown operations at RBS. Shutdown Operations Protection Plan and Primary-Secondary Containment Integrity procedures presently include guidance for closure of the containment hatch and other significant openings in containment, in addition to the requirements contained in the license and design basis. This additional protection will enhance the ability to limit offsite effects.

Acceptance limits for the fuel handling accident are provided in 10 CFR 100 with additional guidance provided in NUREG[–]0800, Section 15.7.4. The proposed changes continue to ensure that the whole-body and thyroid doses at the exclusion area and low population zone boundaries, as well as control room doses, are below the corresponding regulatory limits. These margins are unchanged, therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The commission has provided guidance concerning the application of the standards of 10 CFR 50.92 by providing certain examples (51 FR 7751, March 6, 1986) of amendments that are not considered likely to involve a significant hazards consideration. This proposed amendment is very similar to example (vi):

(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

As we have shown in the preceding discussion, this refinement to the FHA dose calculation results in a small increase to the consequences of a previously analyzed accident, but the results of the change remain clearly within the guidelines of 10 CFR 100, Appendix A, GDC[–]19, and the guidance of SRP [S]ection 15.7.4, without reducing a margin of safety.

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: December 20, 1999.

Description of amendment request: The proposed amendment would change River Bend Station (RBS) Technical Specification (TS) 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' to allow the Inclined Fuel Transfer System (IFTS) primary containment isolation blind flange to be removed during MODE 1, 2, or 3. In its application, the RBS licensee stated that, with the blind flange removed and certain restrictions and administrative controls in place, the IFTS penetration would not represent an uncontrolled breach of the containment boundary and that the containment isolation function would continue to be provided through implementation of these additional controls.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change permits removal of the blind flange on the Inclined Fuel Transfer System (IFTS) when primary containment operability is required in MODE[S] 1, 2, and 3. This will permit operation of the IFTS while the plant is operating. With respect to the probability of an accident, this aspect of the containment structure does not directly interface with the reactor coolant pressure boundary. The removal of this blind flange does not involve modifications to plant systems or design parameters that could contribute to the initiation of any accidents previously evaluated. Operation of IFTS is unrelated to the operation of the reactor, and there is no aspect of IFTS operation that could lead to or contribute to the probability of occurrence of an accident previously evaluated. Removal of the blind flange and operation of IFTS does not result in changes to procedures that could impact the occurrence of an accident.

With respect to the issue of consequences of an accident, the function of the containment is to mitigate the radiological consequences of a loss of coolant accident (LOCA) or other postulated events that could result in radiation being released from the fuel inside containment. While the proposed change does not change the plant design, it does permit alteration of the containment boundary for the IFTS penetration. Altering the containment boundary in this case (*i.e.,* removing the blind flange) results in some IFTS components possibly being subjected to containment pressure in the event of a LOCA. However, the additional post-accident peak pressure load to be imposed upon the components in the IFTS if the blind flange is removed is a small fraction of their design capability. Therefore, they are considered an acceptable barrier to prevent uncontrolled release of post-accident fission products for this proposed change.

The proposed change required examination of two potential leakage pathways. The larger is the IFTS transfer tube, itself. The other, much smaller one, is a branch line used for draining the IFTS transfer tube during its operation. It is clear that the gate valve at the bottom of the transfer tube is always water sealed and maintained so by the submergence of the water in the transfer tube and in the fuel building spent fuel storage pool (the lower pool). The height of this water seal is greater than that necessary to prevent leakage from the bottom of the transfer tube during accidents that result in the calculated peak post-DBA [design basis accident] LOCA pressure, P_a. Furthermore, the hydraulically operated gate valve in the lower end of the tube will remain closed, and has pressure retaining capability greater than that of the containment structure itself. The potential leakage pathway from the drain piping which attaches to the transfer tube will be isolated if required, via administrative controls on the drain piping isolation valve. Additionally, the drain piping isolation valve will be added to the Primary Containment Leakage Rate Testing Program (Technical Specification 5.5.13) to ensure that leakage past this valve will be maintained consistent with the leakage rate assumptions of the accident analysis. Due to the test methodology, the portion of the large transfer tube piping outboard of the blind flange (the portion of the tube which becomes exposed to the containment atmosphere during the draining portion of the IFTS operation) will also be part of the leakage rate test boundary and will therefore also be tested. Therefore, no unidentified leakage will exist from the piping and components that are outboard of the blind flange, and the leakage rate assumptions of the accident analysis will be maintained. Note that the bottom gate valve in the IFTS transfer tube will remain closed for this test evolution.

Therefore, the proposed change does not result in a significant increase in the probability of the consequences of previously evaluated accidents, provided the bottom gate valve remains closed during MODE 1, 2, or 3 operation.

2. The proposed changes would not create the possibility of a new or different kind of accident from any previous analyzed.

The proposed change consists of the removal of a passive component which is not part of the primary reactor coolant pressure boundary nor involved in the operation or shutdown of the reactor. Being passive, its presence or absence does not affect any of the parameters or conditions that could contribute to the initiation of any incidents or accidents that are created from a loss of coolant or an insertion of positive reactivity. Realigning the boundary of the primary containment to include portions of the IFTS is also passive in nature and therefore has no influence on, nor does it contribute to the possibility of a new or different kind of incident, accident or malfunction from those previously analyzed. Furthermore, operation of the IFTS is unrelated to the operation of the reactor and there is no mishap in the process that can lead to or contribute to the possibility of losing any coolant from the reactor or introducing the chance for an insertion of positive or negative reactivity, or any other accidents different from and not bounded by those previously evaluated.

Therefore, the proposed change does not result in creating the possibility of a new or different kind of accident from any accident previously evaluated, provided the bottom gate valve remains closed during MODE 1, 2, or 3 operation.

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

The proposed change involves the realignment of the primary containment boundary by removing the blind flange which is a passive component. The margin of safety that has the potential of being impacted by the proposed change involves the dose consequences of postulated accidents which are directly related to potential leakage through the primary containment boundary. The potential leakage pathways due to the proposed change have been reviewed, and leakage can only occur from the administratively controlled IFTS transfer tube drain piping, and from the IFTS transfer tube itself. A dedicated individual will be designated to provide timely isolation of this drain piping during the duration of time when this proposed change is in effect. The conservatively calculated dose which might be received by the designated individual while isolating the drain piping is calculated to be 3.8 rem TEDE [total effective dose equivalent], which remains within the guidelines of General Design Criterion (GDC) 19 (10 CFR 50, Appendix A, Criterion 19). Furthermore, the drain piping isolation valve will be added to the Primary Containment Leakage Rate Testing Program (Technical Specification 5.5.13) to ensure that leakage from the piping and components located outboard of the blind flange will be maintained consistent with the leakage rate assumptions of the accident analysis.

Studies of the capability of the IFTS system to withstand containment pressurization under severe accident conditions have been conducted. These studies conclude that IFTS, including the transfer tube and its valves, has a capability to withstand beyond design basis severe accident containment pressures which is greater than that of the containment structure itself. The RBS Emergency Operating Procedures (EOPs) are based on an ultimate containment failure pressure capability of 53 psig [pounds per square inch—gauge], which represents a margin of safety of 38 psi above the 15 psig containment design pressure. This margin of safety is not impacted with the IFTS blind flange removed as long as the IFTS bottom valve remains closed. This capability to withstand containment pressurization under severe accident conditions envelops other non-DBA LOCA scenarios, such as the small break LOCA. For the large break LOCA, additional defense-in-depth is provided by maintaining a water seal greater than P_a above the outlet of the IFTS transfer tube in the lower pool.

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

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 15, 1999 (NPF–38–216).

Description of amendment request: One proposed change adds a Technical Specification (TS) Bases Control Program to the Waterford 3 TS Administrative Controls Section, modeled after the guidelines contained in NUREG–1432. Additionally, the proposed change corrects an editorial error identified in the TS following issuance of Amendment 146, dated October 19, 1998.

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

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

Response: The proposed changes to the Waterford 3 [Waterford Steam Electric Station, Unit 3] Technical Specifications add a TS Bases Control Program and correctly reference the appropriate document where administrative controls were relocated. The TS Bases Control Program will provide administrative controls that ensure changes to the TS Bases are appropriately reviewed and consistent with the Updated Final Safety Analysis Report (UFSAR). The addition of the proposed program does not affect any accident initiator or mitigation of any events analyzed in Chapter 15 of the UFSAR. Also, neither change has any affect on the operation of any structures, systems, or components or the assumptions of any accident analyses.

The TS Bases Control Program will ensure that any change to the Bases that involves an unreviewed safety question will receive prior Nuclear Regulatory Commission approval. Changing the reference to the Quality Assurance Program Manual (QAPM) for the item relocated to the QAPM is purely administrative.

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

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

Response: The proposed changes to the Waterford 3 TS add a TS Bases Control Program and correctly reference the appropriate document where administrative controls were relocated. The addition of a TS Bases Control Program represents an administrative function performed under existing regulatory controls consistent with 10 CFR 50.59. The proposed change to reference the appropriate document where an administrative control was relocated is purely administrative in nature. The change merely corrects the Technical Specifications wording to reflect the actual location of the record retention requirements for records of reviews performed on changes to the Process Control Plan (PCP) and Offsite Dose Calculation Manual (ODCM) in the QAPM.

These proposed changes do not involve a change in plant design or affect the configuration or operation of any structure, system, or component, nor does it involve any potential initiating events that would create any new or different kind of accident. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The proposed changes to the Waterford 3 TS add a TS Bases Control Program and correctly reference the appropriate document where administrative controls were relocated. The addition of a TS Bases Control Program is an administrative change and has no [a]ffect on a margin of safety, as defined by Section 2 of the TS. The only [a]ffect of the TS Bases Control Program is to establish controls over how TS Bases changes are reviewed and implemented consistent with 10 CFR 50.59.

The proposed change to a reference in the Administrative Controls section merely corrects the TS wording to reflect the actual location of the record retention requirements for records of reviews performed on changes to the PCP and ODCM in the QAPM.

These proposed changes do not involve a change in plant design or have any affect on the plant protective barriers. Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 15, 1999 (NPF–38–217).

Description of amendment request: The proposed change creates a new Technical Specification (TS) for the Main Feedwater Isolation Valves Section modeled after the guidelines of TS 3.7.3 in NUREG–1432. Additionally, the letter provides for Nuclear Regulatory Commission (NRC) Staff review of an unreviewed safety question regarding the crediting of the Reactor Trip Override feature and Auxiliary Feedwater Pump high discharge pressure trip as assisting the operation of the Main Feedwater Isolation Valves during their required safety function, to close on a Main Steam Isolation Signal.

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

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

Response: The proposed change to add the Main Feedwater Isolation Valves (MFIVs) to the Technical Specifications (TS) and provide an allowed outage time of 72 hours with appropriate required ACTIONs does not affect the operation of any structures, systems, or components or the assumptions of any accident analyses. The MFIVs are primarily designed to mitigate the consequences of a Main Steam Line Break (MSLB), and the Feedwater Line Break (FWLB). This TS change ensures the 5 second closure time currently assumed in the Waterford 3 [Waterford Steam Electric Station, Unit 3] analysis, thus it preserves the current analysis. Hence, the consequences of accidents previously evaluated do not change. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated. Adding the MFIVs to the TS will not initiate an accident. Providing a TS and allowed outage time makes no changes to the plant and, thus, no increase in the probability of any accident previously evaluated.

The accidents/events that may be affected by the proposed resolution to credit the Reactor Trip Override (RTO) circuitry for the Steam Generator [SG] Feed Pumps (SGFPs) during SGFP operation and the crediting of the Auxiliary Feedwater (AFW) pump high discharge pressure trip during AFW pump operation are the MSLB and the FWLB.

The crediting of the RTO circuitry for the SGFPs and the crediting of the AFW pump trip will not affect the probability of occurrence of a MSLB or FWLB. Neither the SGFPs nor the AFW pump are initiators of either line break.

The crediting of the RTO circuitry for the SGFPs and the crediting of the AFW pump trip will not adversely affect the consequences of a MSLB or FWLB. Ultimately, the RTO feature allows more reliable MFIV closure by reducing the differential pressure against which the MFIVs must close while not introducing a new failure mechanism such as a Loss of Feedwater or water hammer event.

The RTO feature (which has always been a part of the Waterford 3 plant design) mitigates the consequences of the MSLB and MFLB by reducing flow to the affected steam generator and containment.

The Loss of Feedwater Event can be initiated by the loss of a SGFP. The currently analyzed Loss of Feedwater Event evaluates the loss of both SGFPs, which bounds a potential loss of one SGFP. Therefore, any modification that could increase the probability of a pump trip could increase the probability of this event. Since the proposed solution of crediting RTO features of the SGFPs and the trip of the AFW pump for the MFIV margin issue uses existing functions, no new features/trips will be added, and there is no increase in the probability or consequences of a Loss of Feedwater Event. The only plant modification being made is to enhance RTO such that it will run the SGFPs back to a minimum speed on a reactor trip, even when the FWCS [Feedwater Control System] is in manual. Although this slows the pump down, feedwater and the SGFPs remain available and the Loss of Feedwater Event probability is not significantly increased. The modification to make RTO function when the FWCS is in manual is not significant since the FWCS is in manual such a short period of time during plant operation.

The AFW system is not credited in any accident analysis. The Emergency Feedwater (EFW) system is relied upon in the safety analyses to replenish SG inventory. Therefore, crediting the AFW pump discharge pressure trip will not involve an increase in the probability or consequences of any accident.

In conclusion, the proposed TS change and resolution to the MFIV margin issue will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different type of

accident from any accident previously evaluated?

Response: The proposed TS change in itself does not change the design or configuration of the plant. No new or different equipment is being installed by the TS. No new or different accidents result from the addition of the MFIVs to the TS. Previously performed accident analyses remain valid. The proposed allowed outage time and required actions of the proposed TS do not change the procedural operation of the plant, but specify the requirements for treatment of the MFIVs under the plant TS. Therefore, no new or different type of accident from any accident previously evaluated is created.

No new system interaction is created by crediting the existing RTO and AFW pump trip. Failure to isolate feedwater would require two failures, failure of the RTO or AFW circuitry, in addition to the failure of the Main Feedwater Regulating Valves (MFRVs) and Startup Feedwater Regulating Valves (SFRVs) to close, and is beyond single failure criteria. If the RTO and AFW features were the single failure, then closure of the regulating valves would be credited for MSIS [Main Steam Isolation Signal] isolation since the regulating valves were designed to close against SGFP shutoff head.

RTO and AFW pump trips would not be considered initiators of a MSLB or FWLB, but could be considered initiators of a Loss of Feedwater Event. However, this event is bounded by the analyzed Waterford 3 Loss of Feedwater Events. No new event is created. The only hardware change being made is the use of RTO for pump run back when the FWCS is in manual. The existing signal will be used and routed through the same methods as are currently installed, ensuring it will run the pump back appropriately. Therefore, no new system interactions or events are created.

The new method of potential failure that has not previously been evaluated is in the fact that Waterford 3 would now be crediting a non-safety related circuit for closure of the safety related MFIVs. Non-safety features are not normally credited for the proper operation of a safety related component. However, in this case, for the valve to close in the 5 seconds assumed in safety analyses, the RTO and AFW pump trip will be credited. Because this is new, different and not a previously approved allowance, this resolution must be submitted for NRC Staff approval. Entergy believes this resolution is acceptable based on the high degree of reliability of these components.

The system design, as discussed above, does not increase the potential for a Loss of Feedwater Event and current analyses bound all potential accident scenarios. Therefore, the proposed TS change and resolution to the MFIV margin issue will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The MFIVs have no [a]ffect on a margin of safety as defined by Section 2 of the TS. Their only [a]ffect is response to the accidents described above, which will be enhanced by specifying an allowed outage time, action requirements and surveillance requirements in the TS. Therefore, no reduction in the margin of safety is involved with the addition of these valves to the TS.

No new system interaction is created by the crediting of the RTO feature or the AFW pump trip, or the addition of RTO operation in manual.

The proposed resolution does affect a part of a protective boundary, the MFIV, which serves to isolate the Main Feedwater system from portions of the system inside containment. However, it does not affect operation or function of the valve itself since no changes to the valve are being made. The proposal allows increased margin for valve closure; therefore, margins of safety are not affected. The valve will close within the time limits required by safety analyses and general design criteria.

Therefore, the proposed TS change and resolution to the MFIV margin issue will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 15, 1999 (NPF–38–218).

Description of amendment request: The proposed changes extend the Reactor Coolant System Pressure Temperature Curves to 20 Effective Full Power Years.

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

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

Response: The proposed changes will not increase the probability or consequences of any accident previously evaluated since the proposed changes revise the pressure/temperature limits in accordance with 10 CFR 50, Appendix G, utilizing the latest NRC [Nuclear Regulatory Commission] guidelines in Regulatory Guide 1.99, Revision 2, relative to estimating neutron irradiation damage to the reactor vessel. The proposed changes also maintain the conservative limits with respect to the low temperature overprotection (LTOP) system and heatup and cooldown restrictions.

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

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

Response: The proposed changes will not create the possibility of a new or different kind of accident from any previously analyzed since they do not introduce new systems, failure modes, or other plant perturbations. The proposed changes revise the pressure/temperature limits in accordance with 10 CFR 50, Appendix G, utilizing the latest NRC guidelines in Regulatory Guide 1.99, Revision 2, relative to estimating neutron irradiation damage to the reactor vessel.

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

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

Response: The proposed changes will not involve a significant reduction in the margin of safety since equal or more stringent pressure/temperature limitation requirements for reactor operation will be applied. The proposed changes were derived in accordance with approved NRC methodology which was developed to assure the reactor coolant system pressure boundary is designed with sufficient margin to withstand any condition during normal operation including anticipated operational occurrences and system inservice leak and hydrostatic tests.

These requirements were revised in accordance with 10 CFR 50, Appendix G, utilizing the latest NRC guidance in Regulatory Guide 1.99, Revision 2, relative to estimating neutron irradiation damage to the reactor vessel. The LTOP system limits were also reanalyzed for the proposed changes.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 19, 1999. (NPF–38–219).

Description of amendment request: The proposed changes modify Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specification (TS) 4.5.2.f.2 by increasing the performance requirement for the low pressure safety injection (LPSI) pumps. The change revises the LPSI pump Surveillance Requirements to measure pump developed head, instead of pump discharge pressure. The associated changes to TS Bases are included in the submittal.

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

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

Response: Increasing the LPSI pump performance requirements will not increase the probability or consequences of any accidents. There are no physical changes to the pump. The only procedure changes required are to Surveillance Procedure OP–903–030, ''Safety Injection Pump Operability Evaluation.'' The changes do not impact plant operating procedures. The LPSI system is primarily designed to mitigate the consequences of a large break Loss of Coolant Accident (LOCA). These proposed changes do not affect any of the assumptions used in the deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not change.

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

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

Response: The proposed change does not alter plant operations, nor does it alter the physical plant. The change only increases existing equipment performance requirements. No different accidents result from the increase in performance requirements. No change is being made to the parameters within which the plant is operated. The setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change will only increase the performance requirements of the LPSI pumps.

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

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

Response: To the contrary, the change increases LPSI pump performance requirements, increasing the margin between the TS performance requirements and the analytical limit.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 4, 1999 (NPF–38–222).

Description of amendment request: The proposed change modifies Technical Specifications (TS) 3.5.2 to extend the allowed outage time (AOT) to seven days for one high pressure safety injection (HPSI) train inoperable and TS 3.5.3 to change the end-state to HOT SHUTDOWN with at least one OPERABLE shutdown cooling train in operation. Additionally, an AOT of 72 hours in TS 3.5.2 is imposed for other conditions where the equivalent of 100 percent emergency core cooling system (ECCS) subsystem flow is available. If 100 percent ECCS flow is unavailable due to two inoperable HPSI trains, an ACTION has been added to restore at least one HPSI to OPERABLE status within one hour or place the plant in HOT STANDBY in six hours and to exit the MODE of applicability in the following six hours. In the event the equivalent of 100 percent ECCS subsystem flow is not available due to other conditions, TS 3.0.3 is entered. The Limiting Condition for Operation terminology is being changed for consistency with the ECCS requirements. Additionally, the associated TS Bases are 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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: The High Pressure Safety Injection System (HPSI) is part of the Emergency Core Cooling System subsystem. Inoperable HPSI components are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The HPSI system is primarily designed to mitigate the consequences of a Loss of Coolant Accident (LOCA). These proposed changes do not affect any of the assumptions used in the deterministic LOCA analyses. Hence the consequences of accidents previously evaluated do not change.

In order to fully evaluate the HPSI AOT extension, probabilistic safety assessment (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. These analyses are detailed in report CE NPSD–1041, ''Joint Applications Report for High Pressure Safety Injection System Technical Specification Modifications,'' March 1998.

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.5.2 does not affect the probability or the consequences of an accident.

The proposed change allows a combination of equipment from redundant trains to be inoperable provided that at least the equivalent of a single ECCS subsystem remains operable. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Allowing equipment from redundant trains to constitute a single operable subsystem does not increase the probability that a failure leading to an analyzed event will occur. The ECCS components are passive until an actuation signal is generated. This change does not increase the failure probability of the ECCS components. This change reduces the plant's susceptibility to common cause failures. As such, the probability of occurrence for a previously analyzed accident are not significantly increased.

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

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

Response: The proposed change does not change the design or configuration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, and the setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change will only provide the plant some flexibility in maintaining the minimum equipment required to be operable to perform the ECCS function while in this condition. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: The proposed changes do not affect the limiting conditions for operation or their bases used in the deterministic analysis to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes involve no significant increase in risk. These evaluations are detailed in report CE NPSD–1041. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed change. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating a transient event. The proposed change, which allows operation to continue for up to 72 hours with components inoperable in both ECCS subsystems, is acceptable based on the remaining ECCS components providing 100% of the required ECCS flow. The reduced potential for a self-induced plant transient resulting from unit shutdown required for a second inoperable ECCS train is minimized. Therefore, the change does not involve a significant reduction in the margin of safety, and is offset by minimizing the potential for a self induced plant transient.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 4, 1999 (NPF–38–223).

Description of amendment request: The proposed change modifies Technical Specification (TS) 3.5.2 to extend the allowed outage time (AOT) to seven days for one low pressure safety injection (LPSI) train inoperable. Additionally, an AOT of 72 hours is imposed for other conditions where the equivalent of 100 percent emergency core cooling system (ECCS) subsystem flow is available. If 100 percent ECCS flow is unavailable due to two inoperable LPSI trains, an ACTION has been added to restore at least one LPSI train to OPERABLE status within one hour or place the plant in HOT STANDBY in six hours and to exit the MODE of applicability in the following six hours. In the event the equivalent of 100 percent ECCS subsystem flow is not available due to other conditions, TS 3.0.3 is entered. The Limiting Condition for Operation terminology is being changed for consistency with the ECCS requirements. Additionally, the associated TS Bases are 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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No. The Low Pressure Safety Injection System (LPSI) is part of the Emergency Core Cooling System subsystem. Inoperable LPSI components are not accident initiators in any accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The LPSI system is primarily designed to mitigate the consequences of a large Loss of Coolant Accident (LOCA). These proposed changes do not affect any of the assumptions used in the deterministic LOCA analysis. Hence, the consequences of accidents previously evaluated do not change.

In order to fully evaluate the LPSI AOT extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequency. As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD–995, Combustion Engineering Owners Group ''Joint Applications Report for Low Pressure Safety Injection System AOT Extension.''

The Configuration Risk Management Program is an Administrative Program that assesses risk based on plant status. Adding the requirement to implement this program for Technical Specification 3.5.2 does not affect the probability or the consequences of an accident.

The proposed change allows a combination of equipment from redundant trains to be inoperable provided that at least the equivalent of single train of ECCS remains operable. Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. Allowing equipment from redundant trains to constitute a single operable train does not increase the probability that a failure leading to an analyzed event will occur. The ECCS components are passive until an actuation signal is generated. This change does not increase the failure probability of the ECCS components. This change reduces the plant's susceptibility to common cause failures. As such, the probability of occurrence for a previously analyzed accident are not significantly increased.

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

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

Response: No. The proposed change does not change the design or configuration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated, and the setpoints at which protective or mitigative actions are initiated are unaffected by this change. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to respond to an off-normal event. As such, no new failure modes are being introduced. The proposed change will only provide the plant some flexibility in maintaining the minimum equipment required to be operable to perform the ECCS function while in this Condition. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No. The proposed changes do not affect the limiting conditions for operation or their bases used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrate that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD–995. The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. None of these are adversely impacted by the proposed change. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating a transient event. The proposed change, which allows operation to continue for up to 72 hours with components inoperable in both ECCS trains, is acceptable based on the remaining ECCS components providing 100% of the required ECCS flow. The reduced potential for a self-induced plant transient resulting from unit shutdown required for a second inoperable ECCS train is minimized. Therefore, the change does not involve a significant reduction in the margin of safety, and is offset by minimizing the potential for a self induced plant transient.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment requests: December 3, 1998.

Description of amendment requests: The proposed amendments would add a new Technical Specification (T/S) and associated Bases for the distributed ignition system (DIS). The proposed change incorporates the technical requirements of NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse 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 amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not involve a significant increase in the probability or consequences of an accident previously evaluated. The amendments involve new requirements for the T/Ss and do not delete any existing requirements.

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

The T/S being proposed for the DIS is consistent with its design and operation as previously reviewed and approved, and therefore, does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The T/S being proposed for the DIS is consistent with [the] design and operation as previously reviewed and approved, and therefore, does not involve a significant reduction in a margin of safety. Compliance with the proposed T/S will provide additional assurance of system availability to maintain a margin of safety for containment integrity during degraded core events.

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: December 15, 1999.

Description of amendment request: This proposed technical specification (TS) change will revise the average power range Monitors (APRMs) neutron flux-high (flow biased) allowable value based on a revised power to flow map. The revised power to flow map extends the current plant operating domain to above the rated rod line, to within an envelope referred to as the maximum extended load line limit (MELLL) and adds the increased core flow (105%) region. The current power to flow map is based on a region bounded by the extended load line limit (ELLL) and evaluations prepared as part of the Core Operating Limits Report (COLR).

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. Attachment 3 [to the December 15,1999 application] (Reference 1) evaluates operation in the Maximum Extended Load Line Limit (MELLL) and Increased Core Flow (ICF) regions and the impact on equipment and safety system performance. Impacts on containment, the reactor vessel, Recirculation System, reactor vessel internals, limiting transients for the Cycle 20 reload (upcoming refuel outage), Loss of Coolant Accident (LOCA), and Anticipated Transients Without SCRAM (ATWS) events were evaluated. The conclusion is that for all events, accidents, and equipment evaluated, operation and event response remain within previously established design limits and acceptance criteria. No changes in the initiators of accidents previously evaluated are being made by this change. Because operation in the expanded regions maintains adequate design margin and there are no changes in the accident initiators, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

In support of operation in the MELLL region, the proposed change modifies (increases) the Average Power Range Monitor (APRM) Neutron Flux-High (Flow Biased) allowable value. Changes to the setpoint and allowable value will be implemented in accordance with approved setpoint methodology and plant procedures (References 7 and 8). As noted in Technical Specifications (TS) Bases Section B.3.3.1.1.2.b: ''No specific safety analyses take credit for the APRM Neutron Flux-High (Flow Biased) Function.'' The APRM allowable value credited in accident analyses is based on the 120% fixed scram-allowable value (TS Table 3.3.1.1–1, Function 2.c), which remains unchanged as a result of this requested TS change. Though not credited in analyses, the limiting flow biased value of 119% Reactor Thermal Power (RTP) also remains unchanged. Evaluations presented in Attachment 3 demonstrate that operation in the MELLL envelope, with reliance on the credited fixed scram allowable value (analytically assumed at 123% RTP to justify a 120% TS allowable value), results in event and accident responses within design limits and established acceptance criteria. Therefore, no significant increase in source term, radiological consequences or other accident consequences occurs as a result of the proposed change.

The proposed change has no affect on operation in the ICF region. The allowable value, as part of the proposed change, will reach its clamped upper limit value of 119% reactor thermal power. Core flows at or above this level will result in the allowable value reaching its current TS upper limit of 119%. As stated above, the limiting value remains unchanged as part of this request.

The postulated failure mechanisms for the equipment are not changed, nor are any design limits exceeded. The proposed change will result in the need to replace APRM equipment to allow operation in the extended power to flow domain. These replacements will be evaluated per the requirements of 10 CFR 50.59 as part of the Cooper Nuclear Station (CNS) design change process to confirm no Unreviewed Safety Question is created. Therefore, implementation of this proposed TS amendment will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change will not create the possibility of a new or different kind of accident than previously evaluated.

This proposed change does not modify the functional requirements of the affected equipment, create any new system interfaces or interactions, create any new process conditions that exceed design limits, nor create any new system failure modes or sequences of events that could lead to an accident.

The postulated failure mechanisms for the equipment are not changed, nor are any design limits or acceptance criteria exceeded. The proposed change will result in the need to replace APRM equipment to allow operation in the extended power to flow domain. These replacements will be evaluated per the requirements of 10 CFR 50.59 as part of the CNS design change process to confirm no Unreviewed Safety Question is created. 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 will not involve a significant reduction in a margin of safety.

Change to the APRM Neutron Flux-High (Flow Biased) allowable value is still limited by the 119% RTP value of TS. This value is not credited in the safety analyses. In addition, the existing 120% fixed scram allowable value (TS Table 3.3.1.1–1, Function 2.c) still provides the same margin to the Analytical Limit of 123% RTP. Analyses documented in Attachment 3 demonstrate that for operation in the MELLL envelope or ICF region, adequate margin to design limits is maintained and event acceptance criteria are met. Thus, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 22, 1999.

Description of amendment request: The proposed license amendment requests Nuclear Regulatory Commission (NRC) review and approval of revisions to the Cooper Nuclear Station (CNS) design basis accident (DBA) radiological assessment calculational methodology used to demonstrate compliance with the Exclusion Area Boundary and Low Population Zone dose acceptance criteria specified in 10 CFR 100.11, and the control room dose acceptance criteria discussed in General Design Criteria (GDC) 19 of 10 CFR 50, Appendix A. The revisions entail a complete rewrite of the radiological assessment calculational methodology. The proposed changes do not revise the accident category, general accident description, identification of accident cause, frequency classification, starting conditions of the accident, accident sequence of events, or system operation as described in the CNS Updated Safety Analysis Report (USAR). The revised radiological assessment calculational methodology does, however, involve changes to the radiological consequence summary, fission product release from fuel assumptions, fission product release to secondary containment assumptions and conditions, fission product release to the environs assumptions and initial conditions, and radiological effects summary described in the CNS USAR. Additionally, the revised CNS DBA radiological assessment calculational methodology incorporates the GDC 19 control room dose acceptance criteria determination as part of the assessment. Previously the control room dose assessment was maintained as separate design calculations and not included in the CNS USAR DBA radiological assessment summaries.

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

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

The proposed revisions to the Design Basis Accident (DBA) radiological assessment calculational methodology do not affect the accident initiators or precursors of accidents previously evaluated. The proposed revisions to the methodology do not affect the existing design, function or operation of systems, structures or components in the facility. No new or different type of plant equipment is installed by the revised radiological assessment calculational methodology. Plant operating modes are not changed due to the proposed revision to the DBA radiological assessment calculational methodology. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the Technical Information Document (TID–14844) source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. The revisions utilize conservatively lower accident mitigation system filter efficiency assumptions and incorporate plant specific accident mitigation system operating parameter and design assumptions which result in a calculated radiological consequence increase. Operation of accident mitigation systems, structures and components is not altered by the changes in accident mitigation assumptions. Due to the broad changes in the calculational methodology and assumptions, and an increase in the postulated accident source term, the calculated radiological dose consequences of each design basis accident have changed and in some cases increased. In each case, however, the calculated radiological dose consequences satisfy the Exclusion Area Boundary and Low Population Zone radiological dose acceptance criteria specified in 10 CFR 100 and the control room dose acceptance criteria discussed in General Design Criteria 19 (GDC 19) of 10 CFR 50, Appendix A. Therefore, the proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed revisions to the DBA radiological assessment calculational methodology do not change the existing design, function or operation of systems, structures or components in the facility. No new or different type of plant equipment is installed by this change. There are no changes to existing design parameters governing plant operation, plant operating modes, or changes in system interfaces. No new types of accident initiators or precursors are created by the proposed revision to the DBA radiological assessment calculational methodology. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the TID–14844 source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. Therefore, the proposed change does not create the possibility of a new or different kind of accident previously evaluated.

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

The proposed revisions to the DBA radiological assessment calculational methodology do not involve a relaxation in the criteria used to establish safety limits or a relaxation in the limiting conditions for operation. The accident analysis sequence of events remains unchanged. The proposed change will not result in any challenges to plant equipment, fuel integrity, or the reactor coolant system pressure boundary. The proposed revisions are calculational in nature and serve only to incorporate more recent site specific meteorological data, reflect plant specific system operating parameters and design, utilize more widely accepted accident assumptions for a facility of Cooper Nuclear Station's vintage, incorporate the TID–14844 source term to be consistent with the accident assumptions used, update fuel parameter considerations to include higher burnup fuel designs, and to utilize generic and updated calculational and software methodologies to perform the analysis. These revisions improve the consistency between the accident dose calculation assumptions and improve the documentation basis for each accident calculation. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 19, 1999.

Description of amendment request: The licensee proposes to change the Technical Specifications (TS) by relocating the specific requirements of TS 6.4.3, ''Nuclear Safety Audit Review Committee (NSARC),'' to the Quality Assurance Program located in the Updated Final Safety Analysis Report.

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 does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed change does not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). The proposed change is administrative in nature and does not decrease the effectiveness of programmatic controls or the procedural details of assuring operation of the facility in a safe manner.

The relocation of the Nuclear Safety Audit Review Committee requirements from the Technical Specification to a new Appendix 17C in UFSAR Chapter 17.2 does not alter the performance or frequency of these activities. Future changes to the Quality Assurance Program are subject to the 10 CFR 50.54(a) and 10 CFR 50.59 and change processes.

The proposed change will not degrade the ability of systems, structures and components important to safety to perform their safety function. The proposed change will not change the response of any system, structure or component important to safety as described in the UFSAR. Since the plant response to an accident will not change, there is no change in the potential for an increase in the consequences of an accident previously analyzed. As such, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed change. The proposed change has no adverse impact on component or system interactions. The proposed change will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the UFSAR. The proposed change is administrative in nature and does not change the level of programmatic controls and procedural details of assuring operation of the facility in a safe manner. The proposed changes involve the relocation of the requirements of the Nuclear Safety Audit Review Committee from TS 6.4.3 to Updated Final Safety Analysis Report, Chapter 17.2, ''Quality Assurance Program'' in a new Appendix 17C. Future changes to the Quality Assurance Program are subject to the 10 CFR 50.54(a) and 10 CFR 50.59 and change processes.

Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes involve the relocation of the requirements of the Nuclear Safety Audit Review Committee from TS 6.4.3 to Updated Final Safety Analysis Report, Chapter 17.2, ''Quality Assurance Program'' in a new Appendix 17C. There is no adverse impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed change is administrative in nature and does not change the level of programmatic controls and procedural details controls of assuring operation of the facility in a safe manner.

Future changes to the Quality Assurance Program are subject to the 10 CFR 50.54(a) and 10 CFR 50.59 change processes. Therefore, relocation of the requirements contained in TS 6.4.3 to the Update Final Safety Analysis Report does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 29, 1999.

Description of amendment request: The licensee proposes to change Technical Specification (TS) Surveillance Requirement (SR) 4.8.1.1.2f., to relocate sub requirement 4.8.1.1.2f.1 which requires inspection of the emergency diesel generators (EDGs) on an 18-month cycle to be subjected to an inspection in accordance with manufacturers recommendations, to the Seabrook Station Technical Requirements Manual (SSTRM).

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 does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated. The proposed change does not alter or prevent the ability of structures, systems and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR).

Performance of EDG inspection activities based on condition-based maintenance rather than time-directed maintenance will neither exacerbate nor significantly increase the probability or consequences of an accident previously evaluated in the Seabrook Station UFSAR. North Atlantic has extensive experience and expertise in operating and maintaining the EDGs to determine the appropriate maintenance activities for demonstrating operability of the EDGs. North Atlantic will continue to use, in conjunction with manufacturer recommendations, prudent engineering judgment when conducting testing, preventive and corrective maintenance activities on the EDGs. In addition, the other surveillance testing required by SR 4.8.1.1.2f would continue to ensure that the EDGs are capable of performing their safety function.

Throughout the first six fuel cycles, overall EDG condition has steadily improved with the use of improved design, utilization of better condition monitoring tools and procedures and the reduction of intrusive preventative maintenance tasks made possible by the improved on-line condition monitoring methods. These improvements resolved problems that were recognized during the early years of EDG operation.

North Atlantic has implemented the Maintenance Rule Program in accordance with the provisions of 10 CFR 50.65, Regulatory Guide (RG) 1.160, and NUMARC 93–01, ''Industry Guide for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.''

North Atlantic's maintenance rule program establishes specific performance criteria for SSCs. Reliability and unavailability performance criteria have been assigned to risk significant and standby safety-related non-risk significant SSCs. Other in-scope SSCs have been assigned appropriate reliability and/or plant level performance criteria. SSCs that are determined to not meet the established performance criteria are designated as (a)(1) and are subject to action plans, goal setting, and goal monitoring. Performance of (a)(1) SSCs is compared to the established goals. When it is determined that the performance goals have been achieved, a SSC may be returned to the normal performance monitoring (a)(2) status.

With regard to the EDGs, these components and the associated support systems are risk significant and standby safety-related. The experience to date, applying the Maintenance Rule Program to the EDGs, has proven to be positive. Risk informed decision-making concerning the benefits of maintenance and time out of service has maintained reliable EDGs with unavailability consistent with the assumptions in the Seabrook Station Probabilistic Risk Assessment (PRA).

Furthermore, Operations Department personnel perform daily, weekly, biweekly, monthly and quarterly walkdowns and inspections of various items as well as the monthly surveillance run on each diesel. These inspections, combined with system control panel alarms, engine oil sampling and on-line monitoring of engine vibration and running performance (cylinder firing, fuel delivery and exhaust temperatures), enable expeditious response to a developing degraded condition and provide a mechanism for failure identification prior to performance of the refueling interval surveillances.

Based on the reviews of the surveillance tests, inspections and maintenance activities, it is concluded that there is no significant impact on the reliability of the EDGs and, therefore, there is no significant increase in the probability or consequences of any previously analyzed accident.

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

The proposed change does not alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed change. The proposed change has no adverse affect on component or system interactions. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated, the proposed change does not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change does not adversely affect equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed change does not adversely affect the EDG's ability to ensure that sufficient power is available to supply the safety related equipment required for: 1) the safe shutdown of the facility, and 2) the mitigation and control of accident conditions within the facility.

Surveillance testing of the EDGs during normal plant operation provides assurance that the proposed change will not adversely affect the reliability of the EDGs. North Atlantic will continue to use, in conjunction with manufacturer's recommendations, prudent engineering judgment when conducting testing, preventive, and corrective maintenance activities on the EDGs. In addition, the other surveillance testing required by SR 4.8.1.1.2f would continue to ensure that the EDGs are capable of performing their safety function. Thus, it is concluded that the EDGs would continue to be available upon demand to mitigate the consequences of an accident and, therefore, there is no significant reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities.Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 3, 1999.

Description of amendment request: The licensee proposes to change the technical specifications (TS) by incorporating reference to the American Society for Testing and Materials (ASTM) Standard D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Charcoal,'' as the test protocol for charcoal filter laboratory testing. In addition, there will be a change to Surveillance Requirements 4.7.6.1d.5) and 4.9.12d.4) specifying a minimum required heater output based on design rated voltage.

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 changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR).

The proposed changes modify the Technical Specifications to reference appropriate test parameters for performing laboratory testing of nuclear-grade charcoal in ESF [engineered safety feature] filtration systems in accordance with ASTM D3803–89. The testing methodology associated with ASTM D3803–89 provides more stringent requirements than what is currently employed. These more stringent requirements will not result in operations that will increase the probability of initiating an analyzed event and 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.

The proposed change associated with verification of heater capacity dissipation by specifying a minimum required output based on design rated voltage does not affect continued operability of the heater. Stipulating the design rated voltage ensures the heater(s) remains capable of performing its safety function. Specifying an upper kW range band is restrictive and has been determined to be unnecessary. There is no safety concern with the heaters operating at a higher kW output. Operating at a higher kW output improves dehumidification. Should maximum operating bus voltage conditions be experienced it does not pose a fire hazard or dry-out concern for the charcoal filters.

There are no changes to previous accident analyses. The radiological consequences associated with these analyses remain unchanged. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not alter the design assumptions, conditions or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes have no impact on component or system interactions.

The proposed changes modify the Technical Specifications to reference appropriate test parameters for performing laboratory testing of nuclear-grade charcoal in ESF filtration systems in accordance with ASTM D3803–89. The changes do impose different, more conservative testing requirements, on the ESF filtration systems charcoal samples. However, there is no alteration in the methods employed to obtain the charcoal sample and testing is performed offsite.

The proposed change associated with verification of heater capacity dissipation by specifying a minimum required output based on design rated voltage does not affect continued operability of the heater. The design function of the heater for humidity control remains unchanged. Deletion of the upper kW range does not pose a fire or dry-out concern for the charcoal filters.

These changes are consistent with the safety analyses and licensing basis. The proposed changes do not introduce any new modes of plant operation, or alter any operational setpoints.

Since the proposed changes do not involve the physical alteration of SSCs (i.e., no new or different type of equipment to be installed) or changes in the methods governing normal plant operation, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

There is no impact on equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The proposed changes modify the Technical Specifications to reference appropriate test parameters for performing laboratory testing of nuclear-grade charcoal in ESF filtration systems in accordance with ASTM D3803–89. The imposition of the more conservative charcoal filter testing requirements associated with ASTM D3803–89 has no significant impact on a margin of safety. The conservative nature of ASTM D3803–89 is by definition, providing additional restrictions to enhance plant safety.

The proposed change associated with specifying a minimum required heater output based on design rated voltage does not reduce the ability of the heater to provide the minimum required kW output for humidity control. Deletion of the upper kW range does not pose a fire or dry-out concern for the charcoal filters.

The proposed changes maintain requirements within the safety analysis and licensing basis. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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: September 7, 1999.

Description of amendment request: The proposed changes affect Technical Specification 3/4.7.8, ''Plant Systems, Snubbers,'' by removing the current special exception which precludes applying the eighteen month functional testing surveillance to the Steam Generator Hydraulic Snubbers.

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 snubbers provide a restraint function to mitigate the consequences of a Main Steam Line Break (MSLB) or to limit seismic induced movements of the steam generators so as to protect the attached Reactor Coolant System (RCS) piping and therefore prevent the initiation of a Loss of Coolant Accident (LOCA).

While the proposed surveillance changes will extend the time period required for 100% inspection of all steam generator snubbers and also the actual service life of the snubber seals, the testing of samples at reduced intervals will actually provide a more reliable and timely indication of snubber functionality and provide increased assurance that generic concerns associated with this snubber set will be detected prior to any failure. The proposed surveillance requirements are the same as currently used for the balance of Millstone Unit No. 2 hydraulic snubbers. Given the complete similarity of design and operation for these components, the sampling approach is well suited for these snubbers. Given the general acceptance of a 10% sampling approach in the general snubber population, its use here for this homogenous set of components is fully justified. In addition to the 10% sample that will be functionally tested on an eighteen month interval, a concurrent 100% visual inspection is conducted during each test period, providing added assurance that no seal failures will go undetected for any significant period. This visual inspection program is unchanged from the existing surveillance program as currently documented in the Millstone Unit No. 2 Technical Specification. The anticipated reliability under the new surveillance frequency and testing methods proposed for the steam generator snubbers will not affect the probability of occurrence of a LOCA or a MSLB as the snubbers' ability to perform their function will prevent over stressing of either the Main Steam (MS) or RCS piping attached to the steam generators. Furthermore, the anticipated reliability under the new surveillance frequency and testing methods proposed for the steam generator snubbers will ensure that the existing evaluated consequences for these accidents will not be increased. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change to Bases Section 3/4.7.8 will delete the text associated with the current exception taken for steam generator snubbers. This change will make the discussion in the Bases consistent with the proposed Technical Specification changes. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes do not alter how any structure, system, or component functions. There will be no effect on equipment important to safety. The proposed changes have no effect on any of the design basis accidents previously evaluated. Therefore, this License Amendment Request does not impact the probability of an accident previously evaluated, nor does it involve a significant increase in the consequences of an accident previously evaluated.

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

The only accidents possible due to failure of the steam generator snubbers to operate properly is increased stresses on both the MS and RCS piping attached to the steam generator due to either additional constraint in the case of premature lockup, or lack of proper constraint in the case of failure to lock-up under dynamic loading. Since the worst case scenario of such a failure would be the initiation of a LOCA, which is currently evaluated in the SAR [safety analysis report], there is no possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will allow use of the preferred approach to snubber surveillance which is in effect for the balance of Millstone Unit No. 2 snubbers. The steam generator snubbers have been previously exempt from the standard approach to snubber surveillance due to the difficulty previously encountered in testing these large and inaccessible components. Given the reliability of these snubbers is not expected to change in that the same requirements as for all other hydraulic snubbers will now consistently be met, there is no significant reduction in a margin of safety. The proposed changes will not alter any of the assumptions used in the accident analysis, nor will they cause any safety system parameters to exceed their acceptance limit. The proposed changes will not affect any operability requirements for equipment important to plant safety. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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: November 23, 1999.

Description of amendment request: The proposed changes will update the list of documents describing the analytical methods used to determine the core operating limits, specified in Technical Specification 6.9.1.8b.

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 document 1 of Technical Specification 6.9.1.8b is made to provide the most recent, Nuclear Regulatory Commission (NRC) approved, methodology description and benchmarking results of the reactor analysis system used in the core neutronics analysis of cycle 14 and beyond. This change has no impact on plant equipment operation. Since the change only affects the neutronics analysis of the core, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change in document 8 of Technical Specification 6.9.1.8b is made to include the most recent, NRC approved, Emergency Core Cooling System (ECCS) model used in Large Break Loss of Coolant Accident (LBLOCA) applications. This model contains resolution of the deficiencies reported under 10 CFR 50.46(a) in a letter dated May 20, 1999. The use of the revised methodology also constitutes an improvement over the previous methodology. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed changes in document 4 of Technical Specification 6.9.1.8b are administrative in nature. Therefore, these changes 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 document 1 of Technical Specification 6.9.1.8b is made to provide the most recent, NRC approved, methodology description and benchmarking results of the reactor analysis system used in the neutronics analysis of cycle 14 and beyond. The proposed change in document 1 of Technical Specification 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 proposed change in the documents in number 8 of Technical Specification 6.9.1.8b is made to include the most recent, NRC approved, ECCS model used in LBLOCA applications. The proposed change in document 8 of Technical Specification 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 proposed changes in document 4 of Technical Specification 6.9.1.8b are administrative in nature. These changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated.

These 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 document 1 of Technical Specification 6.9.1.8b is made to provide the most recent, NRC approved, methodology description and benchmarking results of the reactor analysis system used in the neutronics analysis of cycle 14 and beyond. It has no impact on plant equipment operation. The proposed change in document 8 of Technical Specification 6.9.1.8b is made to include the most recent, NRC approved, ECCS model used in LBLOCA applications. This model contains resolution of the deficiencies reported under 10 CFR 50.46(a) in a letter dated May 20, 1999. The use of the revised methodology still provides a conservative simulation of the LBLOCA and conservative core neutronics analysis. The use of the revised methodology also constitutes an improvement over the previous methodology. The new documents 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 proposed changes in document 4 of Technical Specification 6.9.1.8b are administrative in nature. Therefore, the proposed changes will not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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: December 6, 1999.

Description of amendment request: The proposed changes will modify the Technical Specification (TS) surveillance requirements associated with ensuring a limited number of charging and high pressure safety injection pumps are capable of injecting into the Reactor Coolant System when the plant is shutdown. In addition, the TS Bases will be modified to address these 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. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed modifications to the surveillance requirements (SRs) associated with Technical Specifications 3.1.2.3, 3.1.2.4, and 3.4.9.3 will remove information that specifies the methods to be used to perform the associated SRs. These SRs verify the maximum number of charging and high pressure safety injection (HPSI) pumps capable of injecting into the RCS [Reactor Coolant System] when the plant is shut down. This information will be transferred to the associated Bases. Additional methods associated with the charging pumps, which are technically equivalent to the current method, will be included in the Bases change. This will not change the requirement to verify that the associated pumps are not capable of injecting into the RCS when the plant is shut down.

The proposed changes to the Technical Specifications and Bases will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed changes can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification and Bases changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not significantly alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is unaffected by the changes. 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 modifications to the surveillance requirements associated with Technical Specifications 3.1.2.3, 3.1.2.4, and 3.4.9.3 will remove information that specifies the methods to be used to perform the associated surveillance requirements. This will not change the requirement to verify that the associated pumps are not capable of injecting into the RCS when the plant is shut down.

The proposed changes to the Technical Specifications and Bases will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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: December 7, 1999.

Description of amendment request: The proposed changes to the Technical Specifications (TSs) are associated with the action requirement to suspend positive reactivity additions. These changes will remove the action requirement to suspend positive reactivity additions from TS 3.4.2.1, ''Reactor Coolant System—Safety Valves,'' 3.4.2.2, ''Reactor Coolant System—Safety Valves,'' and 3.7.6.1, ''Plant Systems—Control Room Emergency Ventilation System,'' and provide guidance in the Bases for other TSs that require the suspension of positive reactivity addition.

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.

Technical Specifications 3.4.2.1 and 3.4.2.2

The proposed changes to Technical Specifications 3.4.2.1 and 3.4.2.2, which address the pressurizer code safety valves in Modes 1 through 4, will combine these two specifications into one Technical Specification, 3.4.2. The slight reduction in the Mode of Applicability for the new Technical Specification, to be consistent with the Mode of Applicability for Technical Specification 3.4.9.3, which addresses the Low Temperature Overpressure Protection (LTOP) System, is too small to result in a change in plant operations. The LCO [limiting condition for operation] for the pressurizer code safety valves in Mode 4 with all Reactor Coolant System (RCS) cold leg temperatures > 275 °F will be expanded to require all pressurizer code safety valves to be operable, instead of at least one pressurizer code safety valve. This more restrictive change will require additional accident mitigation equipment to be operable. The proposed action requirements for plant operation in Modes 1, 2, and 3 have been expanded to require the plant to be in Mode 3 within 6 hours and in Mode 4 within the following 6 hours, instead of just Mode 4 within 12 hours. In addition, the action requirements will be modified to address 2 inoperable pressurizer code safety valves. An entry into Technical Specification 3.0.3 will no longer be necessary if both pressurizer code safety valves are inoperable. In addition, the proposed action requirements are more restrictive than the action requirements of Technical Specification 3.0.3. The proposed action requirements for Mode 4 with all RCS cold leg temperatures > 275 °F are different. The new Mode 4 action requirements will direct the plant to be cooled down to the applicability of Technical Specification 3.4.9.3, which will require the LTOP System to be placed in service to provide RCS overpressure protection. The proposed action requirements will ensure that the plant is placed in a condition where sufficient accident mitigation equipment will be available.

The proposed Technical Specification, 3.4.2, will ensure the RCS has adequate overpressure protection when operating above 275 °F. If the pressurizer code safety valves are not operable, the proposed Technical Specification will require a plant shutdown that will place the plant within the capability of the LTOP System to provide RCS overpressure protection. The proposed changes will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed changes can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

Technical Specification 3.7.6.1

The proposed change to Technical Specification 3.7.6.1 will remove the requirement to suspend positive reactivity additions if both control room ventilation trains are inoperable in Modes 5 and 6. The Control Room Ventilation System is required to be operable in Modes 5 and 6 to protect the control room operators from an event that results in a rapid release of radioactivity, such as a fuel handling accident. In Modes 5 and 6, the positive reactivity addition methods of concern are boron dilution, RCS cooldown (negative isothermal temperature coefficient), and control rod withdrawal. Positive reactivity additions associated with fuel handling are already addressed by the additional action requirement in this specification to suspend core alterations. Control rod withdrawal is prohibited by Technical Specification 3.1.3.7, unless the RCS boron concentration is greater than or equal to the refueling boron concentration of Technical Specification 3.9.1. If the RCS is borated to the refueling concentration, sufficient negative reactivity has been added to compensate for the positive reactivity addition associated with control rod withdrawal in Modes 5 and 6. Therefore, only boron dilution and RCS temperature changes are of concern. However, both of these methods will result in slow changes to core reactivity in Modes 5 and 6, and since adequate shutdown margin (SDM) will have been established prior to entering Mode 5 or 6 (Technical Specifications 3.1.1.2 and 3.9.1), neither method will result in a rapid release of radioactivity. Therefore, the requirement to suspend positive reactivity additions is not necessary for the protection of the control room operators.

The proposed change will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed change can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not significantly alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Also, the response of the plant and the operators following these accidents is unaffected by the changes. 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 to combine Technical Specifications 3.4.2.1 and 3.4.2.2 into a new Technical Specification, 3.4.2, will result in a slight reduction in the Mode of Applicability for the new Technical Specification, will require both pressurizer code safety valves to be operable in Mode 4 with all RCS cold leg temperatures > 275 °F, will modify the action requirements in Modes 1, 2, and 3 to add a requirement to be in Mode 3 within 6 hours and to address two inoperable pressurizer code safety valves, and will provide different action requirements for Mode 4 with all RCS cold leg temperatures > 275 °F. The reduction in Mode of Applicability is too small to adversely impact plant operations. Requiring both pressurizer code safety valves to be operable in Mode 4 with all RCS cold leg temperatures > 275 °F will provide additional accident mitigation equipment. The modified action requirement to be in Mode 3 within 6 hours will not change the requirement to be in Mode 4 within 12 hours. The action requirements added to address two inoperable pressurizer code safety valves are more restrictive than the action requirements of Technical Specification 3.0.3. The new Mode 4 action requirements will direct the plant to be cooled down to the applicability of Technical Specification 3.4.9.3, which will require the LTOP System to be placed in service to provide RCS overpressure protection. The proposed action requirements will ensure that the plant is placed in a condition where sufficient accident mitigation equipment will be available.

The proposed change to Technical Specification 3.7.6.1 will remove the requirement to suspend positive reactivity additions if both control room ventilation trains are inoperable in Modes 5 and 6. The Control Room Ventilation System is required to be operable in Modes 5 and 6 to protect the control room operators from an event that results in a rapid release of radioactivity, such as a fuel handling accident. The proposed change will only impact slow methods to change core reactivity, such as boron dilution and RCS temperature changes. Therefore, the action requirement to suspend positive reactivity additions is not necessary for the protection of the control room operators.

The proposed changes will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Northeast Nuclear Energy Company, et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: November 23, 1999.

Description of amendment request: The proposed change affects Technical Specification 4.0.5, ''Limiting Conditions for Operation and Surveillance Requirements'' by adding a biennial or 2-year surveillance interval and incorporating a required frequency for performing inservice testing activities of once per 731 days.

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

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

The proposed change amends Technical Specification Section 4.0.5.b by adding a biennial or 2 year surveillance to the existing list. This surveillance interval is included as part of the current Millstone Unit Nos. 2 and 3 Inservice Test (IST) surveillance program. Inclusion of this surveillance interval in the facility Technical Specifications clarifies the applicability of this surveillance interval and affords operational flexibility in the event a surveillance cannot be completed within the required interval.

The proposed change will have no adverse effect on plant operation, or the availability or operation of any accident mitigation equipment. The plant response to the design basis accidents will not change. In addition, the proposed change can not cause an accident. Therefore, there will be no significant increase in the probability or consequences of an accident previously evaluated.

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

The biennial surveillance relates to performing inservice testing of plant components. The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed Technical Specification change does not introduce a new mode of plant operations and does not involve physical modifications to the plant. Therefore, the proposed change 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.

There is no impact on the margin of safety as defined in the Technical Specifications. Performance of surveillance tests at regular intervals provides assurance of reliability and availability of accident mitigating equipment. The Technical Specifications provide the required frequency for performing surveillance testing. Adding a new surveillance frequency to the Technical Specifications will provide consistent yet acceptable flexibility in scheduling surveillance tests and provide additional assurance that testing will be performed in a timely manner.

The proposed change will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis. Therefore, there will be no significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 29, 1999.

Description of amendment request: The requested changes would revise Technical Specification (TS) 3/4.6.6, ''Supplementary Leak Collection and Release System,'' (SLCRS), TS 3/4.7.7, ''Control Room Emergency Ventilation System,'' (CREVS), TS 3/4.7.9, ''Auxiliary Building Filter System,'' (ABFS), and 3/4.9.12, ''Fuel Building Exhaust System,'' (FBES), in response to Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The requested changes require testing of nuclear-grade activated charcoal to be conducted in accordance with American Society for Testing Materials (ASTM) D3803-1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' as recommended by GL 99–02.

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:

NNECO [Northeast Nuclear Energy Company] has reviewed the proposed revision in accordance with 10 CFR 50.92 and has concluded that the revision does not involve any Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not satisfied. The proposed TS revision does not involve an SHC because the revision would not:

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

The proposed change modifies the TS to reference ASTM D3803–[19]89 for performing laboratory testing of nuclear-grade charcoal in ESF [Engineered Safeguards Features] filtration systems. The testing methodology associated with ASTM D3803–[19]89 provides more stringent requirements than what is currently employed. These more stringent requirements, along with a factor of safety of greater than or equal to two in regards to the charcoal efficiency assumed in the design bases dose analysis will not result in operations that will increase the probability of initiating an analyzed event and 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. There are no related modifications to any systems. The proposed change does not affect procedures governing plant operations. Therefore there is no significant increase in the probability [or consequences] of occurrence of a previously evaluated accident.

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

The proposed change modifies the TS to reference ASTM D3803–[19]89 for performing laboratory testing of nuclear-grade charcoal in ESF filtration systems. The proposed change does not involve the physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. This change does impose different, more conservative testing requirements on the ESF filtration system charcoal samples. However there is no alteration in the methods employed to obtain the charcoal sample and testing is performed offsite. These changes are consistent with the safety analyses and licensing basis. Furthermore, the proposed changes do not introduce any new modes of plant operation, or alter any operational setpoints. Thus the possibility of a new or different kind of accident from any previously evaluated is not created.

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

The proposed change modifies the TS to reference ASTM D3803–[19]89 for performing laboratory testing of nuclear-grade charcoal in ESF filtration systems. The imposition of the more conservative charcoal filter testing requirements associated with ASTM D3803–[19]89 along with a factor of safety of greater than or equal to two, in regards to the charcoal efficiency assumed in the design bases dose analysis has no impact on, nor decreases the margin of plant safety. The conservative nature of ASTM D3803–[19]89 is by definition, providing additional restrictions to enhance plant safety. This change maintains requirements within the safety analysis and licensing basis. Therefore, there will be no significant reduction in the margin of safety as defined in the Bases for the TS affected by the proposed change.

As described above this TSCR [Technical Specification Change Request] does not impact the probability of an accident previously evaluated, does not involve a significant increase in the consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not result in a significant reduction in a margin of safety. Therefore, NNECO has concluded that the proposed changes do not involve an SHC.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 5, 1999.

Description of amendment request: The proposed changes will revise LGS Technical Specifications (TSs) to incorporate revised testing and acceptance criteria for the performance of laboratory analysis of safety-related nuclear-grade activated charcoal in response to Generic Letter (GL) 99–02. ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999. In addition, minor editorial changes are being proposed for wording consistency and to correct a typographical error.

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

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

Changing the methodology for the performance of the laboratory testing of nuclear-grade activated charcoal samples from Reg. [Regulatory] Guide 1.52 to ASTM D3803–1989 in accordance with Generic Letter 99–02, and establishing a new methyl iodide penetration acceptance criteria does not involve any physical changes or modifications to the function or operation of any safety-related structure, system, or component. The new testing methodology will enable a more accurate, conservative and reliable determination of the charcoal decontamination efficiencies associated with the SGTS [Standby Gas Treatment System], RERS [Reactor Enclosure Recirculation System], and CREFAS [Control Room Emergency Fresh Air System] which will better assure that the assumed charcoal efficiencies credited in the licensed accident analysis are adequately maintained. Implementing this change will only involve revisions to existing procedures.

The SGTS, RERS, and CREFAS are standby systems that are designed to mitigate the consequences of the analyzed accidents. No analyzed accident initiating events are impacted, no new accident initiators or new failure modes are created and the credited charcoal efficiency for each system in the licensed accident analyses is not changing as a result of the proposed changes. The ability of the SGTS, RERS, and CREFAS to perform all of their safety-related mitigation functions as designed will not be affected by the proposed changes. Furthermore, the change in the testing methodology and acceptance criteria will not result in increasing the dose rates currently calculated in the existing accident analyses.

In addition, the proposed minor editorial changes are administrative in nature and do not impact the operation, physical configuration, or function of plant equipment or systems. The proposed editorial changes do not impact the initiators or assumptions of analyzed events, nor do they impact mitigation of accidents or transient events.

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

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

Changing the methodology for the performance of the laboratory testing of nuclear-grade activated charcoal in accordance with Generic Letter 99–02, and establishing new methyl iodide penetration acceptance criteria is not an accident initiator, does not create any new failure modes, nor does it result in the occurrence of an accident. This change does not result in any physical plant modification and does not affect the safety-related function, assigned charcoal efficiency assumed in the accident analyses, or operation of the SGTS, RERS, and CREFAS. This change will only involve revisions to existing procedures.

In addition, the proposed minor editorial changes are administrative in nature and do not alter plant configuration, require that new equipment be installed, alter assumptions made about accidents previously evaluated, or impact the operation or function of plant equipment.

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

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

The safety-related air cleaning units used in ESF [Engineered Safety Feature] ventilation systems reduce the potential onsite and offsite consequences of a radiological accident by adsorbing radioiodine. Changing the methodology for the performance of the laboratory testing of nuclear-grade activated charcoal samples from Reg. Guide 1.52 to ASTM D3803–1989 in accordance with Generic Letter 99–02, and the establishment of new methyl iodide penetration acceptance criteria does not increase the dose rates above what is currently calculated in the accident analyses.

In addition, the proposed minor editorial changes are administrative in nature and do not involve any physical changes to plant structures, systems or components (SCCs), or the manner in which SSCs are operated, maintained, modified, tested, or inspected. The proposed editorial changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions of operation, or design parameters for any SSC. The proposed editorial changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of application for amendments: November 17, 1999.

Description of amendment request: The proposed changes will revise the Peach Bottom Units 2 and 3 Technical Specifications (TSs) Section 5.5.7.c., Ventilation Filter Testing Program (VFTP), in accordance with Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' This TS change will (1) specify that the laboratory testing for methyl iodide penetration be performed referencing ASTM D3803–1989 at a temperature of 30 °C (86 °F), and (2) revise the acceptance criteria for methyl iodide penetration.

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.

Changing the methodology for the performance of the laboratory testing of nuclear grade activated charcoal samples from RG [Regulatory Guide] 1.52 to ASTM [American Society for Testing and Materials] D3803–1989 and the establishment of new methyl iodide penetration acceptance criteria and test temperature in accordance with Generic Letter 99–02, do not involve any changes or modifications to the function or operation of any safety related structure, system, or component. The new testing methodology enables a more accurate and conservative charcoal decontamination efficiency to be determined which better assures that the assumed charcoal efficiency credited in the licensed accident analysis is being adequately maintained. Implementing this change only involves revisions to existing procedures.

The SGTS [Standby Gas Treatment System] and MCREVS [Main Control Room Emergency Ventilation System] are standby systems that are designed to mitigate the consequences of the analyzed accidents. No analyzed accident initiating events are impacted, no new accident initiators or new failure modes are created and the credited charcoal efficiency for each system in the licensed accident analyses is not changing. The change in laboratory testing methodology does not degrade the ability of these systems to perform all of their safety related mitigation functions as designed.

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

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

Changing the methodology for the performance of the laboratory testing of nuclear grade activated charcoal in accordance with Generic Letter 99–02 and establishing new methyl iodide penetration acceptance criteria is not an accident initiator, does not create any new failure modes, nor does it result in the occurrence of an accident. This change does not result in any physical plant modification and does not affect the safety related function, charcoal efficiency, or operation of the SGTS or MCREVS. This change only involves revisions to existing procedures to comply with NRC guidance from GL 99–02.

Therefore, the possibility of a new or different kind of accident than previously evaluated is not created.

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

The safety related air cleaning units used in ESF [Engineered Safety Feature] ventilation systems reduce the potential onsite and offsite consequences of a radiological accident by absorbing radioiodine. Changing the methodology for the performance of the laboratory testing of nuclear-grade activated charcoal samples from RG 1.52 to ASTM D3803–1989 in accordance with Generic Letter 99–02, and the establishment of new methyl iodide penetration acceptance criteria does not increase the dose rates above what is currently calculated in the accident analyses.

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

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

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

NRC Section Chief: James W. Clifford.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: November 16, 1999.

Description of amendment request: The proposed amendment would revise the Trojan Nuclear Plant (TNP) Permanently Defueled Technical Specifications by removing Figure 4.1–1, ''Site and Exclusion Area Boundaries,'' from Section 4.0, ''Design Features,'' and incorporate the applicable portion of this figure in the Trojan Nuclear Plant Defueled Safety Analysis Report. Other associated administrative changes resulting from the deletion of Figure 4.1–1, as well as an editorial change to the table of contents, are also proposed.

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

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

The requested license amendment consists of changes that are administrative and/or editorial in nature, in that the physical and operational characteristics of the TNP site are unchanged. As such, the requested amendment does not in any way affect systems, structures, or components that could initiate or be required to mitigate the consequences of an accident previously evaluated. 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 requested license amendment consists of changes that are administrative and/or editorial in nature, in that the physical and operational characteristics of the TNP site are unchanged. As such, the requested amendment does not affect systems, structures, or components in any way not previously evaluated, and no new or different failure modes will be created. 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 requested license amendment consists of changes that are administrative and/or editorial in nature, in that the physical and operational characteristics of the TNP site are unchanged. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas R. Nichols, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Michael T. Masnik.

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

Date of amendment request: December 27, 1999.

Description of amendment request: The proposed amendment would revise Technical Specifications (TS) 4.6.2.2.b, ''Suppression Pool Spray,'' and 4.6.2.3.b, ''Suppression Pool Cooling,'' to modify the acceptance criteria associated with flow rate testing of the Residual Heat Removal (RHR) system pumps.

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 TS change does not involve any physical changes to plant structures, systems or components (SSC). The RHR system will continue to function as designed. The RHR system is designed to mitigate the consequences of an accident, and therefore, cannot contribute to the initiation of any accident. The proposed TS surveillance requirement changes implement testing methods that more appropriately control and reflect RHR operation and establish acceptance criteria, which ensure that Hope Creek's licensing and design basis assumptions are met. In addition, this proposed TS change will not increase the probability of occurrence of a malfunction of any plant equipment important to safety, since the manner in which the RHR system is operated is not affected by these proposed changes. The proposed surveillance requirement acceptance criteria ensure that the RHR safety functions will be accomplished. Therefore, the proposed TS changes would not result in the increase of the consequences of an accident previously evaluated, nor do they involve an increase in the probability 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 TS changes do not involve any physical changes to the design of any plant SSC. The design and operation of the RHR system is not changed from that currently described in Hope Creek's licensing basis. The RHR system will continue to function as designed to mitigate the consequences of an accident. Implementing the proposed changes does not result in plant operation in a configuration that would create a different type of malfunction to the RHR system than any previously evaluated. In addition, the proposed TS changes do not alter the conclusions described in Hope Creek's licensing basis regarding the safety related functions of this system.

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

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

The proposed changes contained in this submittal would implement testing methods that adequately demonstrate RHR pump capability and establish acceptance criteria consistent with Hope Creek's licensing basis. The ability of RHR to perform its safety functions is not adversely affected by these proposed changes. Therefore, the proposed TS change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 29, 1999.

Description of amendment request: The proposed amendments would revise the Salem Nuclear Generating Station Technical Specification requirements for instrumentation in the reactor trip system by adding tolerances to certain setpoint values.

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

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

The accidents of concern affected by the over-temperature or over-power delta temperature [trip signal] which have been evaluated are unaffected by the proposed editorial changes thus the changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

The changes proposed are editorial in nature and do not alter physical configuration, replace or modify existing equipment, affect operating practices or create any new or different accident precursors which could impact on the accident analysis. Thus there is no possibility of a new or different kind of accident as a result of the proposed changes.

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

No margin of safety will be reduced by the proposed changes. The proposed changes do not adversely affect the ability of the trip systems to operate when called upon. Rather, these changes should result in clarity regarding the proper calibration of the trip instrumentation and therefore the margin of safety is preserved for those events in which there is a dependence upon an over-temperature or over-power delta temperature trip signal.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 30, 1999.

Description of amendment request: The proposed amendment would allow the Rochester Gas and Electric Corporation to revise Sections 5.5.10 (a.3), (c.5), and (d.3) of the Ginna Station Improved Technical Specifications (ITS) to provide a reference to American Society for Testing and Materials (ASTM) Standard Procedure D3803–1989 as the procedure for performing laboratory testing of charcoal adsorbers that are installed in the Ginna Control Room Emergency Air Treatment System (CREATS), Containment Post-Accident Sampling System (CPASS), and Spent Fuel Pool Charcoal Absorber System (SFPCAS). These charcoal adsorbers for the CREATS and CPASS are installed for the purpose of reducing the levels of radioactive iodide species released to the containment and control room during a postulated design basis, while the charcoal adsorbers in the SFPCAS are installed for reducing the levels of radioactive iodide species released to the auxiliary building during a postulated fuel handling accident. The changes to ITS Sections (a.3), (c.5), and (d.3) will also provide a specific test temperature and humidity level for performing the testing of the charcoal adsorbers, and to increase the allowable penetration of methyl iodide to these systems from 10% to 14.5%. The requests for the changes are consistent with the staff's position stated in NRC Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. With respect to the more restrictive proposals associated with providing a reference to ASTM D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' and providing a specific test temperature and relative humidity for testing the charcoal adsorbers, the proposed changes do not involve a significant hazards consideration as discussed below:

(1) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes add a reference to the latest approved test protocol and provide for specific test conditions. This does not increase the probability of an accident previously evaluated since the tests are of themselves not an accident initiator. The proposed changes are in accordance with NUREG–1431 guidance and provide a higher assurance of the ability of the charcoal adsorbers to perform as assumed in the accident analysis. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes add specific details of charcoal adsorber testing and do not of themselves involve a physical alteration of the plant (*ie.* no new or different type of equipment will be added to perform the required testing) or changes in the methods governing normal plant operation. The changes only involve implementing currently approved test methodology. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only add conservatism in the test requirements for the charcoal adsorbers credited in the accident analysis. ASTM D3803–1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of accurately and consistently determining the capability of the charcoal. Therefore, this change does not involve a significant reduction in a margin of safety.

With respect to the less restrictive proposal to increase the allowable test limit for methyl iodide penetration of charcoal adsorbers, the changes do not involve a significant hazards consideration as discussed below:

(4) Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes revise the acceptance criteria for the allowed penetration of methyl iodide during the testing of charcoal adsorbers in the plant ventilation systems. This does not increase the probability of an accident previously evaluated since the tests are of themselves not an accident initiator. Because ASTM D3803–1989 is a more accurate and demanding test than older tests this new protocol will allow the use a safety factor of 2 for determining the acceptance criteria for charcoal filter efficiency. The new acceptance criteria continue to ensure that the efficiency assumed in the accident analysis is still valid. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

(2) Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes of revising charcoal adsorber testing acceptance criteria do not of themselves involve a physical alteration of the plant (*ie.* no new or different type of equipment will be added to perform the required testing) or changes in the methods governing normal plant operation. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

(3) Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes only revise the test acceptance criteria of charcoal adsorbers as the result of implementing testing in accordance with ASTM D3803–1989. ASTM D3803–1989 is considered to be the most accurate and most realistic protocol for testing charcoal in ventilation systems because it offers the greatest assurance of

accurately and consistently determining the capability of the charcoal. Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the preceding information, the Rochester Gas and Electric Corporation determined that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

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

Attorney for licensee: Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Marsha Gamberoni, Acting.

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

Date of amendment request: September 30, 1999 (TS 98–005).

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant Unit 1 Technical Specifications (TS) analytical methods for core operating limits to implement an analysis supporting a more negative moderator temperature coefficient (MTC) for the end of cycle condition. This alternate methodology is based on a Westinghouse Electric Company analysis documented in reports WCAP–15088–P, Revision 1 (proprietary), ''Safety Evaluation Supporting a More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear plant,'' and WCAP–15099–P, Revision 1 (non-proprietary, same title).

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 more negative EOL [end-of-life] MTC does not increase the probability of an accident previously evaluated in the FSAR [Final Safety Analysis Report]. No new performance requirements are being imposed on any system or component such that any design criteria will be exceeded. The conservative MDC [moderator density coefficient] assumption in the current analyses of record has been confirmed to remain bounding for the more negative proposed TS values. Therefore, no change in the modeling of the accident analysis conditions or response is necessary in order to implement this change. The consequences of an accident previously evaluated in the FSAR are not increased due to the more negative EOL MTC. The dose predictions presented in the FSAR remain valid such that no more severe consequences will result.

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

The more negative EOL MTC does not create the possibility of an accident which is different than any already evaluated in the FSAR. No new failure modes have been defined for any system or component nor has any new limiting single failure been identified. Conservative assumptions for MDC have already been modeled in the FSAR analyses and it has been determined that the more negative MTC values to be implemented in the TS will continue to be bounded by these assumptions.

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

The evaluation of the more negative EOL MTC has taken into account the applicable technical specifications and has bounded the conditions under which the specifications permit operation. The applicable technical specification is Section 5.9.5.b which lists methods approved by the NRC for use in determining the core operating limits. The values of the LCO [limiting condition for operation] and SRs [surveillance requirements] are located in the COLR [core operating limits report]. The analyses which support these technical specifications have been evaluated. The results as presented in the FSAR remain bounding for the more negative EOL MTC. Therefore, the margin of safety, as defined in the bases to these technical specifications, is not reduced.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard Correia.

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

Date of amendment request: December 21, 1999.

Description of amendment request: This proposed change revises the control rod block requirements consistent with the BWR/4 Standard Technical Specifications. Some functions are proposed to be relocated to the Technical Requirements Manual, the requirements for the retained functions are clarified, and two functions are added to the Technical Specifications.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The relocated functions are not assumed as initial conditions for, nor are they credited in the mitigation of, any design basis accident or transient previously evaluated. Since reactor operation with these revised and relocated Specifications is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events nor assumed mitigation of design basis accident or transient events.

More stringent and purely administrative changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

None of the proposed changes affects any parameters or conditions that could contribute to the initiation of an accident. No new accident modes are created since the manner in which the plant is operated is unchanged. No safety-related equipment or safety functions are altered as a result of these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

There is no impact on equipment design or operation, and there are no changes being made to safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. Since the changes have no effect on any safety analysis assumption or initial condition, the margins of safety in the safety analyses are maintained. In addition, neither administrative changes with no technical impact, nor the imposition of more stringent requirements have a negative impact on a margin of safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 15, 1999 (ET 99–0050).

Description of amendment request: The proposed amendment would modify the Improved Technical Specifications (ITSs) that were issued in Amendment No. 123 on March 31, 1999, and implemented on December 18, 1999. The proposed changes would expand the region of acceptable seal injection flow in Figure 3.5.5–1 of ITS 3.5.5 and provide the following 10 editorial changes: (1) delete the redundant ''%'' sign in the allowable value for function 4 in Table 3.3.1–1 on reactor trip system instrumentation, (2) delete the extra spacing in the description of function 20 in Table 3.3.1–1, (3) insert periods at the end of the text for Conditions M and N in the actions for limiting condition for operation (LCO) 3.3.2 on engineered safety features actuation system instrumentation (ESFASI), (4) spell ''requirements'' correctly in function 5.c of Table 3.3.2–1 for ESFASI, (5) delete the unneeded ''SR 3.3.2.6'' from the surveillance requirements column for Function 7.a in Table 3.3.2–1, (6) align the wording ''Coincident with Safety Injection'' with the title of Function 7.b in Table 3.3.2–1, (7) align the data in the 4 columns of Table 3.3.7–1, CREVS [control room emergency ventilation system] Actuation Instrumentation, for Function 3 with the first line of the title of the function, (8) align the specified completion time in Condition B of the actions for LCO 3.7.1 for main steam safety valves with text for the Required Action B.2, (9) add the acronym ''EES'' to Emergency Exhaust System in the table of contents and use the acronym in the upper right-hand-corner of the 4 ITS pages for LCO 3.7.13 on the emergency exhaust system, and (10) uncapitalize the word ''Associated'' in Condition B of the actions for LCO 3.8.4 on DC sources—operating because it should not be capitalized. The licensee would also add text to the Bases to the applicable safety analyses for the seal injection flow of LCO 3.5.5.

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 restriction on RCP [reactor coolant pump] seal injection flow limits the amount of ECCS [emergency core cooling system] flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI [safety injection]. The intent of the LCO 3.5.5 limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS [reactor coolant system] via the injection points. The expansion of the Acceptable Range for the flow limits does not impact the assumed ECCS flow that would be available for injection into the RCS following an accident.

There are no hardware changes nor are there any changes in the method by which any safety related plant system performs its safety function. Since the change continues to ensure 100 percent of the assumed charging flow is available, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed changes do not involve a physical alteration of the plant (no new of different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose any new or eliminate any old requirements. The expansion of the Acceptable Range for the [seal injection] flow limits does not impact the assumed ECCS flow that would be available for injection into the RCS following an accident.

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

Thus, the 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 change [for seal injection flow] does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. The expansion of the Acceptable Range for the flow limits does not impact the assumed ECCS flow that would be available for injection into the RCS following an accident.

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

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

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

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: December 22, 1999.

Brief description of amendments: The amendments would delete the Donald C. Cook (D.C. Cook), Unit 1 and 2, Technical Specification (TS) 5.4.2, ''Reactor Coolant System Volume,'' because the information regarding the reactor coolant system (RCS) is not required by TS Section 5.0, ''Design Features,'' for compliance with 10 CFR 50.36(c)(4). Changes to the RCS volume information are included in the D.C. Cook Updated Final Safety Analyses Report, and are controlled in accordance with 10 CFR 50.59.

*Date of publication of individual notice in **Federal Register***: January 13, 1999 (65 FR 2199).

Expiration date of individual notice: February 14, 2000.

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

Date of amendment request: March 31, 1999, as supplemented by letters dated May 20, June 1, July 14, and October 14, 1999.

Description of amendment request: The amendment converts the current Technical Specifications (TSs) for the James A. FitzPatrick Nuclear Power Plant, to a set of improved TSs based upon NUREG–1433, ''Standard Technical Specifications for General Electric Plants BWR/4'' Revision 1 dated April 1995.

Date of publication of individual notice in **Federal Register**: November 8, 1999 (64 FR 60854).

Expiration date of individual notice: December 8, 1999.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: June 11, 1999.

Brief description of amendment: The amendment made various title changes to the plant organization.

Date of issuance: January 7, 2000.

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

Amendment No.: 219.

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

*Date of initial notice in **Federal Register***: July 14, 1999 (64 FR 38027).

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

No significant hazards consideration comments received: No.

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois

Date of application for amendments: October 2, 1998, as supplemented by letters dated April 13, 1999, and September 15, 1999. Information in Commonwealth Edison correspondence dated July 8, 1999, and August 30, 1999, was also considered during the review of the amendments.

Brief description of amendments: The amendments replace the custom operational technical specifications with a set of permanently defueled technical specifications that reflect the permanently shutdown and defueled status of the Zion Nuclear Power Station, Units 1 and 2. The amendments also delete certain license conditions from the operating licenses that are no longer applicable to the facility in its permanently shutdown and defueled condition. Information supplied in Commonwealth Edison letters dated July 8, 1999, August 30, 1999, and September 15, 1999, provided clarifying information and did not expand the scope of the original **Federal Register** notice dated June 2, 1999, and did not change the staff's proposed no significant hazards finding.

Date of issuance: December 30, 1999.

Effective date: December 30, 1999.

Amendment Nos.: Unit 1—180; Unit 2—167.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications and the operating licenses.

*Date of initial notice in **Federal Register***: June 2, 1999 (64 FR 29709).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 30, 1999.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 10, 1999 (NRC–99–0072), as supplemented November 19, 1999 (NRC–99–0107).

Brief description of amendment: The amendment revises the Technical Specification surveillance requirements for the Division I 130/260-volt dc battery to accommodate the design of the replacement battery.

Date of issuance: January 12, 2000.

Effective date: As of the date of issuance and shall be implemented prior to the startup from the seventh refueling outage.

Amendment No.: 136.

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

*Date of initial notice in **Federal Register***: November 3, 1999 (64 FR 59800). The November 19, 1999, 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 consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 16, 1999, supplemented November 3, 1999.

Brief description of amendments: The amendments revise Section 3.8.4, ''DC Sources—Operating'' of the Technical Specifications. Specifically, the amendments modify Surveillance Requirements (SRs) 3.8.4.8 and 3.8.4.9 and the associated Bases SR 3.8.4.8 and 3.8.4.9 to allow testing of the direct current (dc) channel batteries with the units on line. The change to SR 3.8.4.8 would also prohibit the diesel generator batteries from being service tested while the units are on line.

Date of issuance: January 7, 2000.

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

Amendment Nos.: Unit 1–183; Unit 2–175.

Facility Operating License Nos. NPF–35 and NPF–52: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: October 20, 1999 (64 FR 56529). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 7, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 9, 1999.

Brief description of amendment: This amendment revised the Perry Nuclear Power Plant Environmental Protection Plan by eliminating the requirement to sample Lake Erie sediment in the Perry and Eastlake Plant area for Corbicula, since Corbicula and zebra mussels have already been identified, and control and treatment plans have been implemented which are effective for both species.

Date of issuance: January 5, 2000.

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

Amendment No.: 110.

Facility Operating License No. NPF–58: This amendment revised the Environmental Protection Plan.

Date of initial notice in ***Federal Register***: November 3, 1999 (64 FR 59802). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 5, 2000.

No significant hazards consideration comments received: No.

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: November 3, 1999.

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

Date of issuance: January 6, 2000.

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

Amendment Nos.: 239 and 220.

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

Date of initial notice in ***Federal Register***: December 1, 1999 (64 FR 67335).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 18, 1998, as supplemented by letters dated March 25, September 29, and November 3, 1999.

Brief description of amendments: The amendments change 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 will no longer require ASW and CCW system train separation after the transfer to hot leg recirculation following a LOCA.

Date of issuance: January 13, 2000.

Effective date: January 13, 2000, and shall be implemented in the next periodic update to the FSAR Update in accordance with 10 CFR 50.71(e).

Amendment Nos.: Unit 1—138, Unit 2—138.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Final Safety Analysis Report Update.

Date of initial notice in ***Federal Register***: October 7, 1998 (63 FR 53953).

The supplemental letters dated March 25, September 29, and November 3, 1999, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 13, 2000.

No significant hazards consideration comments received: No.

PP&L, Inc., Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: March 12, 1999, as supplemented by letter dated November 1, 1999.

Brief description of amendment: This amendment revised the Minimum Critical Power Ratio safety limits in TS Section 2.1.1.2 and modified the references in TS Section 5.6.5 of a critical power correlation applicable to Siemens Power Corporation Atrium-10 fuel.

Date of issuance: December 30, 1999.

Effective date: As of date of issuance and shall be implemented upon startup from the Unit 1 eleventh refueling and inspection outage currently scheduled for spring 2000.

Amendment No.: 186.

Facility Operating License No. NPF–14: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: April 7, 1999 (64 FR 17029).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 30, 1999.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: December 1, 1998, as supplemented by your letters of April 21, July 19, October 18, and November 11, 1999.

Brief Description of amendments: The proposed amendments would revise the Technical Specifications to reflect replacing the current Model 51 steam generators with Westinghouse Model

54F steam generators. The replacement program includes re-analyzing and evaluating loss-of-coolant-accident (LOCA) and non-LOCA mass and energy releases, containment and sub-compartment pressure and temperature responses, dose analyses, and the effects on nuclear steam supply and balance of plant systems.

Date of issuance: December 29, 1999.

Effective date: As of the date of issuance and shall be implemented prior to Unit 1 entering Mode 5 for Cycle 17 (Spring 2000) and prior to Unit 2 entering Mode 5 for Cycle 15 (Spring 2001).

Amendment Nos.: Unit 1–147; Unit–238.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Improved Technical Specifications.

*Date of initial notice in **Federal Register:*** October 20, 1999 (64 FR 56533). The supplemental letters dated October 18, and November 11, 1999, provided clarifying information that did not change the initial proposed no significant hazards consideration determinations.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 29, 1999.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 22, 1998, as supplemented by letters dated June 16, October 21 and 27, November 17, and December 9, 1999.

Brief description of amendments: The amendments revised the Technical Specifications to reflect the steam generator water level low-low trip setpoint differences between the existing Model E and the replacement Model Delta-94 steam generators for the reactor trip system and the engineered safety features actuation system instrumentation.

Date of issuance: December 29, 1999.

Effective date: December 29, 1999, to be implemented following replacement of Unit 1 Model E steam generators with Model Delta-94 steam generators and prior to entry into Operational Mode 3.

Amendment Nos.: Unit 1—120; Unit 2—108.

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

*Date of initial notice in **Federal Register:*** September 9, 1998 (63 FR 48268).

The June 16, October 21 and 27, November 17, and December 9, 1999, supplements provided additional clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 29, 1999.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 19th day of January 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–1732 Filed 1–25–00; 8:45 am]

BILLING CODE 7590–01–P

SMALL BUSINESS ADMINISTRATION

[Declaration of Disaster [#3232]]

State of Kentucky

As a result of the President's major disaster declaration on January 10, 2000, I find that Crittenden, Daviess, and Webster Counties in the State of Kentucky constitute a disaster area due to damages caused by tornadoes, severe storms, torrential rains, and flash flooding that occurred on January 3–4, 2000. Applications for loans for physical damage as a result of this disaster may be filed until the close of business on March 10, 2000 and for economic injury until the close of business on October 10, 2000 at the address listed below or other locally announced locations: U.S. Small Business Administration, Disaster Area 2 Office, One Baltimore Place, Suite 300, Atlanta, GA 30308.

In addition, applications for economic injury loans from small businesses located in the following contiguous counties may be filed until the specified date at the above location: Caldwell, Hancock, Henderson, Hopkins, Livingston, Lyon, McLean, Ohio, and Union Counties in Kentucky; Hardin County, Illinois; and Spencer and Warrick Counties in Indiana.

The interest rates are:

For Physical Damage

Homeowners with credit available elsewhere—7.500%

Homeowners without credit available elsewhere—3.750%

Businesses with credit available elsewhere—8.000%

Businesses and non-profit organizations without credit available elsewhere—4.000%

Others (including non-profit organizations) with credit available elsewhere—6.750%

For Economic Injury

Businesses and small agricultural cooperatives without credit available elsewhere—4.000%

The number assigned to this disaster for physical damage is 323212, and for economic injury the numbers are 9G4100 for Kentucky, 9G4200 for Illinois, and 9G4300 for Indiana.

(Catalog of Federal Domestic Assistance Program Nos. 59002 and 59008)

Dated: January 14, 2000.

Herbert L. Mitchell,

Acting Associate Administrator for Disaster Assistance.

[FR Doc. 00–1804 Filed 1–25–00; 8:45 am]

BILLING CODE 8025–01–P

SMALL BUSINESS ADMINISTRATION

Declaration of Economic Injury Disaster #9G20

State of Washington

King, Pierce, Snohomish, Thurston, and Whatcom Counties and the contigous counties of Chelan, Grays Harbor, Island, Kitsap, Kittitas, Lewis, Mason, Okanogan, Skagit, and Yakima in the State of Washington constitute an economic injury disaster area due to the effects of the warm water current known as El Nino beginning in 1997. Eligible small businesses and small agricultural cooperatives without credit available elsewhere may file applications for economic injury assistance for this disaster until the close of business on September 22, 2000 at the address listed below or other locally announced locations: U.S. Small Business Administration, Disaster Area 4 Office, P.O. Box 13795, Sacramento, CA 95853–4795.

The interest rate for eligible small businesses and small agricultural cooperatives is 4 percent.

(Catalog of Federal Domestic Assistance Program No. 59002)

Dated: Dec. 22, 1999.

Aida Alvarez,

Administrator.

[FR Doc. 00–1805 Filed 1–25–00; 8:45 am]

BILLING CODE 8025–01–P