For the Nuclear Regulatory Commission.

Pamela J. Henderson,

Chief Nuclear Materials Safety Branch 1, Division of Nuclear Materials Safety, Region I.

[FR Doc. 03–18544 Filed 7–21–03; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting Notice

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of July 21, 28, August 4, 11, 18, 25, 2003.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of July 21, 2003

There are no meetings scheduled for the Week of July 21, 2003.

Week of July 28, 2003—Tentative

There are no meetings scheduled for the Week of July 28, 2003.

Week of August 4, 2003—Tentative

There are no meetings scheduled for the Week of August 4, 2003.

Week of August 11, 2003—Tentative

There are no meetings scheduled for the Week of August 11, 2003.

Week of August 18, 2003—Tentative

There are no meetings scheduled for the Week of August 18, 2003.

Week of August 25, 2003—Tentative

Wednesday, August 27, 2003

9:30 a.m.—Briefing on License Renewal Program, Power Update Activities, and High Priority Activities (Public Meeting) (Contact: Ho Nieh, 301–415–1721).

This meeting will be webcast live at the Web address—*http://www.nrc.gov.*

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

Additional Information

By a vote of 3-0 on July 16, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Intergovernmental Issues (Closed—Ex. 9)'' be held on July 16, and on less than one week's notice to the public.

The NRC Commission Meeting Schedule can be found on the Internet at: *http://www.nrc.gov/what-we-do/policy-making/schedule.html.*

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to *dkw@nrc.gov.*

Dated: July 17, 2003.

D.L. Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 03–18682 Filed 7–18–03; 10:14 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, June 27, 2003, through July 10, 2003. The last biweekly notice was published on July 8, 2003 (68 FR 40707).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative 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 Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By August 21, 2003, 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* 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: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: June 3, 2003.

Description of amendment request: Pursuant to title 10 of the Code of Federal Regulations (10 CFR), Section 50.90, Duke Energy Corporation requested an amendment to the McGuire Nuclear Station Facility Operating Licenses and Technical Specifications (TS). The proposed change would modify TS 3.6.14 to allow a pressurizer hatch to be open for up to 6 hours, an increase from the current TS limit of 1-hour. Conforming changes would also be made to the associated Bases.

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

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

No. Implementation of this amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Removal of the pressurizer enclosure hatch will not cause an increase in the probability of an accident which has been previously evaluated because the pressurizer enclosure hatch is not an accident initiator.

The consequences of an accident which have been previously evaluated will not be significantly increased by removal of the pressurizer enclosure hatch. As discussed in the analysis contained in the technical justification supporting this amendment request, the new containment compression peak pressure will remain well below the acceptance criteria. Additionally, the long term containment peak pressure will not be adversely affected due to the delay time in melting of the ice. The removal of the pressurizer enclosure hatch itself has been previously evaluated in Modes 1 through 4 in accordance with the analytical process described in NUREG–0612 and the NRC's December 22, 1980 letter regarding the control of heavy loads at nuclear plants. The changes proposed in this license amendment request will have no adverse effect on the procedures used for the handling of heavy loads (pressurizer enclosure hatch) at McGuire nor on the generation of internal missiles as evaluated in Section 3.5 of the McGuire Updated Final Safety Analysis Report.

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

No. Implementation of this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of the NRC approval of this license amendment request. As discussed above, extending the time that the pressurizer hatch is allowed to be open does not create any new or different accidents from those previously evaluated. Removal of the pressurizer enclosure hatch to perform inspections or maintenance inside the pressurizer cavity has been previously evaluated and determined to be acceptable. The analysis contained in the technical justification for this license amendment request provides results which conclude that the containment compression peak pressure, and the long term containment peak pressure are acceptable with the pressurizer enclosure hatch open. This amendment does not impact any plant systems that are accident initiators; therefore, no new accident types are being created.

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

No. Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The pressurizer enclosure hatch and its performance have a direct impact on the containment boundary, since peak containment pressure due to an accident could be affected. However, the analysis supporting this amendment request concludes that the containment compression peak pressure and the long term containment peak pressure continue to be acceptable with the increased open time for the hatch. Thus the performance of the fission product barriers will not be significantly impacted by implementation of this amendment and no safety margin will be significantly impacted.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: June 30, 2003.

Description of amendment request: The proposed amendment would revise the control room emergency ventilation system (CREVS) surveillance requirement (SR) by modifying an existing SR related to the makeup flow rate to show that it is applicable to the VSF–9 train and by adding a new makeup flow rate SR that is applicable to the 2VSF–9 train.

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

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

Response: No.

The purpose of the CREVS is to provide airborne radiological protection for operations from the control room for the design basis loss of coolant accident fission product release and for a fuel handling accident. The proposed change continues to assure that the control room operator will be protected from the dose consequences related to either of these accidents.

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

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

Response: No.

The proposed change will establish appropriate outside air makeup flow rates for the 2VSF–9 fan unit. This criterion has been evaluated and determined to continue to provide protection to the control room operator in accordance with General Design Criterion 19. The proposed change is not an accident initiator. No modifications to the system are proposed which would create the possibility of a new or different kind of accident.

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

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

Response: No.

The proposed change will establish the allowable makeup airflow into the control room when the 2VSF–9 CREVS train is in operation. Calculations have been performed which demonstrate that the proposed flow criteria provides increased protection for the control room operator.

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

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

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 No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 30, 2003.

Description of amendment request: The proposed amendment would: (1) Eliminate credit for the Boraflex neutron absorbing material used for reactivity control in Region 1 of the spent fuel pool (SFP), (2) credit a combination of soluble boron and several defined fuel loading patterns within the storage racks to maintain SFP reactivity within the effective neutron multiplication factor (K_{eff}) limits of 10 CFR 50.68, (3) increase the minimum boron concentration in the SFP to 2000 parts per million (ppm), and (4) reduce the fresh fuel assembly initial enrichment to less than or equal to 4.55 ± 0.05 weight percent uranium-235 (U–235).

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

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

Response: No.

The fuel handling accidents described below can be postulated to increase reactivity. However, for these accident conditions, the double contingency principle of ANS [American Nuclear Society] N16.1–1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

Three types of drop accidents have been considered: a vertical drop accident, a horizontal drop accident, and an inadvertent drop of an assembly between the outside periphery of the rack and the pool wall.

- A vertical drop directly upon a cell will cause damage to the racks in the active fuel region. The proposed 2000 ppm soluble boron concentration will ensure that K_{eff} does not exceed 0.95.
- A fuel assembly dropped on top of the rack that comes to rest horizontally will not deform the rack structure such that criticality assumptions are invalidated. The rack structure is such that an assembly positioned horizontally on top of the rack results in a minimum separation distance from the upper end of the active fuel region of the stored assemblies. This distance is sufficient to preclude interaction between the dropped assembly and the stored fuel.
- An inadvertent drop of an assembly between the outside periphery of the rack and the pool wall is bounded by the worst case fuel misplacement accident condition.

The fuel assembly misplacement accident was considered for all storage configurations. An assembly with high reactivity is assumed to be placed in a storage location which requires a fuel assembly with a lower reactivity. The presence of soluble boron in the pool water assumed in the analysis has been shown to offset the worst case reactivity effect of a misplaced fuel assembly for any configuration. This soluble boron requirement is less than the proposed 2000 ppm that will be required by the ANO–2 [Arkansas Nuclear One, Unit No. 2] TS [Technical Specifications]. Thus, a five percent subcriticality margin can be easily met for postulated accidents, since any reactivity increase will be much less that the negative worth of the dissolved boron.

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

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

Response: No.

The proposed change will define several acceptable 2 x 2 loading patterns and acceptable interfaces between the patterns. In addition, the proposed change will credit soluble boron to assure a five percent subcriticality margin is maintained during normal conditions and in the event of a postulated accident. The soluble boron concentration assumed in the analyses for a postulated accident is less than the proposed TS change of 2000 ppm. Thus, a five percent subcriticality margin can easily be met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

No new or different types of fuel assembly drop scenarios are created by the proposed change. The presence of soluble boron in the SFP water assures a subcriticality margin is maintained in the event of fuel assembly misplacement.

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

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

Response: No.

With the presence of a nominal boron concentration, the fuel storage patterns are designed to assure that fuel assemblies of less than or equal to 4.55 ± 0.05 weight percent U–235 enrichment when loaded in accordance with the proposed loading patterns will be maintained within a subcritical array with a five percent subcritical margin (95% probability at the 95% confidence level). This has been verified by criticality analyses.

Credit for soluble boron in the SFP water is permitted under accident conditions as well as in non-accident conditions. Criticality analyses have been performed to determine the required boron concentration that would ensure a subcriticality margin of at least five percent. By increasing the minimum boron concentration to greater than 2000 ppm, the margin of safety currently defined by taking credit for soluble boron will be maintained.

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

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

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 No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: June 30, 2003.

Description of amendment request: The proposed amendment would (1) reorganize the Arkansas Nuclear One, Unit No. 2 (ANO–2) Technical Specifications (TSs) Section 6.0, Administrative Controls, (2) modify the ANO–2 Facility Operating License, and actions and surveillance requirements (SRs) of various other TSs, to support the reorganization of Section 6.0, and (3) modify several actions and SRs that are related to systems that are shared by ANO–2 and Arkansas Nuclear One, Unit No. 1 (ANO–1). These changes are being proposed so that the philosophy and location of the TSs in Section 6.0 reflect the recently approved conversion of the ANO–1 TSs to the Improved Technical Specifications (ITS) and the subsequent amendments to the ANO–1 ITS. This amendment request supersedes the previous application related to the revision of TS Section 6.0 dated January 31, 2002, as supplemented on June 26 and July 18, 2002. The January 31, 2002, application was previously noticed in the **Federal Register** on March 19, 2002 (67 FR 12602).

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

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

Response: No.

Administrative Changes

The proposed changes involve reformatting and rewording of the existing TSs. The reformatting and rewording process involves no technical changes to existing requirements. As such, the proposed changes are administrative in nature and do not impact initiators of analyzed events or assumed mitigation of accident or transient events.

Less Restrictive—Administrative Deletion of Requirements

The proposed changes relocate requirements from the TSs to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements.

More Restrictive Changes

The proposed changes provide more stringent requirements for the ANO–2 TSs. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The more stringent requirements are imposed to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis and to provide greater consistency with the ANO–1 TS and NUREG 1432.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

The control room area radiation monitor is used to support mitigation of the consequences of an accident; however, it is not considered the initiator of any previously analyzed accident. Also, the addition of the Note to allow time for testing reduces the potential for initiation of a previously analyzed accident due to reduced potential for shutdowns and startups due to incomplete or missed surveillances. As such, the proposed revision to include an allowance for testing does not significantly increase the probability of any accident previously evaluated. This change does not result in any hardware changes, but does allow operation for a limited time with an inoperable monitor for the purposes of testing. Since the capability of the control room area radiation monitor to provide the required information continues to be verified, and the time allowed for inoperability for testing is short, the change will not reduce the capability of required equipment to mitigate the event. Also, the consequences of an event occurring during the proposed operation of the unit during the allowed inoperability for testing are the same as the consequences of an event occurring while operating under the current TS Actions. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the CREVS [control room emergency ventilation system] to be inoperable due to control room boundary inoperability for a period of 24'hours.

Neither CREVS nor the control room boundary is the initiator of any accident analyzed in the SAR [Safety Analysis Report]. Therefore, this change does not result in a significant increase in the probability of an accident previously evaluated.

The CREVS and the control room boundary are intended to provide a habitable environment for the control room operators in the event of an accident that results in the release of radioactivity to the environment. The allowance to open the control room boundary intermittently is acceptable, because of the administrative controls that will be implemented to ensure that the opening can be rapidly closed when the need for control room isolation is indicated, restoring the control room habitability envelope. Allowing both CREVS trains to be inoperable for 24 hours due to an inoperable control room boundary is acceptable because of the low probability of an accident requiring control room isolation during any given 24 hour period, because entry into this condition is expected to be an infrequent occurrence, and because preplanned compensatory measures to protect the control room operators from potential hazards are implemented. Therefore, this change will not result in a significant increase in the probability [consequences] of an accident previously evaluated.

(3) An allowance will be added to allow use of a ''simulated'' or ''actual'' test signal when testing the automatic isolation feature of the control room air filtration system.

The phrase ''actual or simulated'' in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an ''actual'' signal, nor does it eliminate any restriction on producing an ''actual'' signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the function of the system functional test remains unaffected the change does not involve a significant increase in the consequences of an accident previously evaluated.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the individual volume is greater than 17,446 gallons will be added. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

The AC Sources are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR. Equipment powered by the AC Sources, which may be considered as an initiator, continues to be assured of electrical power. The proposed increased restoration time involves parameters unrelated to initiating the failure of the AC Sources. As such the proposed time allowance for restoration of limited levels of readiness parameter degradation will not increase the probability of any accident previously evaluated. The proposed changes allow additional time for restoration of parameters that have been identified as not immediately affecting the capability of the power source to provide its required safety function. The identified parameters are capable of being replenished during operation of the diesel generators, and the short additional allowable action time continues to provide adequate assurance of operable required equipment. Therefore, this change does not involve a significant increase in the probability of or the consequences of any accident previously evaluated.

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

The testing of diesel generator fuel oil is not considered an initiator, or a mitigating factor, in any previously evaluated accident. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is being added. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

This change does not result in any changes in hardware or methods of operation. The change allowing the absence of the STA or the radiation protection technician is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

This change does not result in any changes in hardware or methods of operation. The change in the support relationship between the STA and the control room staff is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

This change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternatives to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of

personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

The controls for access to a high radiation area are not considered as initiators, or as a mitigation factor, in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

The testing of diesel generator fuel oil is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

Notification of the Vice President, Operations ANO when a safety limit is violated is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

This change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(13) A change to frequency of the integrated leak tests for each system outside containment that could contain highly radioactive fluids from ''at a frequency not to exceed refueling cycle intervals'' to ''at least once per 18 months.''

Performance of the integrated leak tests for each system outside containment that could contain highly radioactive fluids is not an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(14) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

The extension of the testing frequency, up to 25% of the test interval, is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or 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. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Administrative Changes

The proposed changes do not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operations. The proposed changes will not impose any different requirements.

Less Restrictive—Administrative Deletion of Requirements

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operations. The proposed changes will not impose any different requirements and adequate control of the information will be maintained.

More Restrictive Changes

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do impose different requirements. However, these changes do not impact the safety analysis and licensing basis.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

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

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the control room ventilation system (CREVS) to be inoperable due to a control room boundary inoperability for a period of 24 hours.

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) An allowance will be added to allow use of a ''simulated'' or ''actual'' test signal when testing the automatic isolation feature of the control room air filtration system.

The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the individual volume is greater than 17,446 gallons will be added. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

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

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is proposed. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

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

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the support relationship the STA provides the control room staff and does not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

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

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternates to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

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

(13) A change to frequency of the integrated leak tests for each system outside containment that could contain highly radioactive fluids from ''at a frequency not to exceed refueling cycle intervals'' to ''at least once per 18 months.''

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(14) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

Administrative Changes

The proposed changes will not reduce the margin of safety because they have no impact on any safety analysis assumptions. The changes are administrative in nature.

Less Restrictive—Administrative Deletion of Requirements

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the TSs to other license basis documents, which are under licensee control, are the same as the existing TSs. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements.

More Restrictive Changes

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

(a) increasing the scope of the specification to include additional plant equipment,

(b) providing additional actions,

(c) decreasing restoration times, or

(d) imposing new surveillances.

The changes are consistent with the safety analysis and licensing basis.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

The margin of safety for the control room area radiation monitor is based on availability and capability of the instrumentation to provide the required information to the operator. The frequency is based on unit operating experience that demonstrates channel failure is rare, and on the use of less formal but more frequent checks of channels during normal operational use of the displays associated with the required channels. Therefore, the availability and capability of the control room area radiation monitor continues to be assured by the proposed Surveillance Requirements and this change does not involve a significant reduction in a margin of safety.

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the control room ventilation system (CREVS) to be inoperable due to control room boundary inoperability for a period of 24 hours.

This change does not involve a significant reduction in a margin of safety since: (1) administrative controls will be in place to ensure that an open control room boundary can be rapidly closed when a need for control room isolation is indicated; and (2) an inoperable control room boundary that renders both trains of CREVS inoperable is an infrequent occurrence, the probability of an accident requiring control room isolation during any given 24 hour period is low, and preplanned compensatory measures to protect the control room operators from potential hazards are implemented.

(3) An allowance will be added to use a simulated or actual test signal when testing the automatic isolation feature of the control room air filtration system.

Use of an actual signal instead of the existing requirement which limits use to a simulated signal, will not affect the performance of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between ''actual'' or ''simulated'' signals. Therefore, the change does not involve a significant reduction in a margin of safety.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the individual volume is greater than 17,446 gallons. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

The parameter limits provide substantial margin to the parameter values that would be absolutely necessary for diesel generator operability. When the parameters are less than their limits this margin is reduced. However, the availability of AC Sources continues to be assured since the allowed time for parameters to be less than their limits is short and the allowed levels for the parameters are adequate to provide the immediately needed power availability. Further, the parameters can be restored to within limits during the proposed time provided should they be required. Therefore, this change does not result in a signification reduction in [a] margin of safety.

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

The proposed change allows the stored diesel fuel oil total particulates to be outside the required limits for seven days before declaring the associated diesel inoperable. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). The seven day allowance provides an appropriate backstop to ensure the particulate level is restored to within limits in a reasonable time period. Since the diesel is still capable of performing its function the margin to safety is not reduced.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is proposed. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

The margin of safety is not dependent on the presence of the STA or the radiation protection technician. Therefore, this change does not involve a significant reduction in a margin of safety.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

The margin of safety is not dependent upon who the STA supports. Therefore, this change does not involve a significant reduction in a margin of safety.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternatives to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

The requirements for control of high radiation areas provide for the use of alternates to the ''control device'' or ''alarm signal'' requirements of 10 CFR 20.1601. This change provides such alternative methods for controlling access. These methods and additional administrative requirements have been determined to provide adequate controls to prevent unauthorized and inadvertent access to such areas. Therefore, this change does not involve a significant reduction in a margin of safety.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

The testing of stored diesel generator fuel oil is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM–D975) once every 31 days. The change reflects industry-standard acceptable DG fuel oil testing programs. Over the storage life of ANO–2 DG fuel oil, the properties tested by ASTM–D975 are not expected to change and performing these tests once on the new fuel oil provides adequate assurance of the proper initial quality of fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability. Therefore, this change does not involve a significant reduction in a margin of safety.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

The margin of safety is not dependent upon notification of the Vice President, Operations ANO upon the violation of a TS safety limit. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(13) A change to frequency of the integrated leak tests for each system outside containment that could contain highly radioactive fluids from ''at a frequency not to exceed refueling cycle intervals'' to ''at least once per 18 months.''

The current and proposed frequencies of this test are equivalent for all practical purposes. Therefore, this change does not involve a significant reduction in a margin of safety.

(14) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

The proposed allowance allows a possible increase in performance interval. However, the test will still be performed at reasonable intervals to ensure the intent of the surveillance is maintained. Therefore, this change does not involve a significant reduction in 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: May 28, 2003, as supplemented on June 24, 2003

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.4.3, ''RCS Pressure and Temperature (P/T) Limits,'' and Section 3.4.12, ''Low Temperature Overpressure Protection (LTOP),'' to incorporate revised reactor pressure vessel P/T limits and overpressure protection system limits to allow operation up to 20 effective full-power years. Specifically, the proposed amendment would revise TS Figures 3.4.3–1 to 3.4.3–3 and TS Figures 3.4.12–1 to 3.4.12–4.

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

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

Response: No.

The probability of occurrence of an accident previously evaluated for Indian Point 3 is not altered by the proposed amendment to the technical specifications (TSs). The accidents remain the same as currently analyzed in Final Safety Analysis Report (FSAR) as a result of changes to the P/T and LTOP limits. The new P/T and LTOP limits were based on the NRC [Nuclear Regulatory Commission] approved, for Indian Point 3, Westinghouse/Combustion Engineering methodology along with American Society of Mechanical Engineers (ASME) Code (Boiler and Pressure Vessel Code) alternatives including Code Case N–640. Code Case N–640 has been accepted for use by the NRC but has not been incorporated into Reg. [Regulatory] Guide 1.147, Rev. 12, at this time. An exemption is being submitted separately for the use of Code Case N–640. The proposed changes do not impact the integrity of the reactor coolant system pressure boundary (RCPB) as a result of this change. In addition there is no increase in the potential for the occurrence of a loss of coolant accident. The probability of any design basis accident is not affected by the change, nor are the consequences of any design basis accident affected by the proposed change. The proposed P/T limit curves and LTOP limits are not considered to be an initiator or contributor to any accident currently evaluated in the Indian Point 3 FSAR. These new limits ensure the long term integrity of the RCPB.

Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. A new reactor vessel specimen was withdrawn at the most recent refueling outage and will be analyzed over the next year to enhance the database used to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head.

The predicted radiation induced ΔRT_{NDT} (shift in reference temperature nil-ductility transition) was calculated using the respective reactor vessel beltline copper and nickel contents and the neutron fluence applicable to normal plant performance through the remainder of the operating license, using the most up-to-date cross sections methodologies, as documented in the recent Appendix K power uprate report.

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

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

Response: No.

The proposed changes to the P/T and the LTOP limits will not create a new accident scenario. The requirements to have P/T and LTOP protection are part of the licensing basis of Indian Point 3. The proposed changes reflect the change in vessel material properties acknowledged and managed by regulation and the best data available in response to NRC Generic Letter 92–01, Revision 1. The approach used meets NRC and ASME regulations and guidelines. The Westinghouse/Combustion Engineering methodology has been approved for use at Indian Point 3 by the NRC. Code Case N–640 has been found acceptable by the NRC to be used at other nuclear plants. By separate letter ENO [Entergy Nuclear Operations, Inc.] is requesting an exemption to use Code Case N–640 because the Code Case has not been incorporated in Regulatory Guide 1.147, Rev. 12, at this time. The adjusted reference temperatures for fracture toughness are consistent with that previously provided to the NRC [* * *] The data analysis for the vessel specimen removed to date, confirm that the vessel materials are responding as predicted.

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

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

Response: No.

The existing P/T curves and LTOP limits in the technical specifications are reaching their expiration period for the number of years at effective full power operation. The revision of the P/T limits and curves will ensure that Indian Point 3 continues to operate within margins allowed by 10 CFR 50.60 and the ASME Code. The material properties used in the analysis are based on results established through Westinghouse/Combustion Engineering material reports for copper and nickel content. The material properties were evaluated in parallel using statistical methodology. The results are consistent and for conservative purposes, the more restrictive result is used. The application of Code Case N–640 presents alternative procedures for calculating P/T and LTOP temperatures and pressures in lieu of that established for ASME Section XI, Appendix G–2215. This Code alternative allows certain assumptions to be conservatively reduced. However, the procedures allowed by Code Case N–640 still provide significant conservatism and ensure an adequate margin of safety in the development of P/T operating and pressure test limits to prevent non-ductile fractures.

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

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: August 16, 2002, as supplemented June 6, 2003.

Description of amendment request: The proposed amendment would add a new Technical Specification (TS) requirement to the Pilgrim Nuclear Power Station (Pilgrim) TSs consistent with Technical Specification Task Force (TSTF)–358, Revision 5. TSTF–358 addresses modifications to requirements for missed surveillances consistent with NUREG 1433, Revision 2, ''Standard Technical Specification, General Electric Plants, BWR/4'' (STS) surveillance requirement (SR) 3.0.3. The proposed amendment to the Pilgrim TSs would be added as TS 4.0.3.

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the model NSHC determination in its application dated August 16, 2002, as supplemented on June 6, 2003.

In addition, the following statement would be added to the TS definition of Limiting Condition for Operation (LCO): ''Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO.'' The proposed amendment would also make administrative changes to add new TS Sections 3.0, ''Limiting Condition for Operation (LCO) Applicability,'' and 4.0, ''Surveillance Requirement (SR) Applicability,'' into the Pilgrim TSs. New TSs 3.0, 4.0.1, and 4.0.2 would be identified as ''Not Used.'' These changes are proposed to rectify the differences in the format and terminology of the current Pilgrim TSs to the STS. The associated Bases would also be implemented.

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

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

[CLIIP Changes]

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

[Additional Changes]

The proposed change involves an addition to clarify the required action when an SR is not met and new TS sections for consistency with the STS. These additions do not involve technical changes to the existing TSs. As such, these changes provide clarity and are administrative in nature and do not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

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

[CLIIP Changes]

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

[Additional Changes]

The proposed change involves an addition to clarify the required action when a SR is not met and new TS sections for consistency with the STS. The changes do not involve physical alterations to the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

[CLIIP Changes]

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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

[Additional Changes]

The proposed change involves an addition to clarify the required action when a SR is not met and new TS sections for consistency with the STS. These additions do not involve technical changes to the existing TSs. The changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. Also, since these changes provide clarity and are administrative in nature, no question of safety is involved. Therefore, there will be no 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. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 4, 2001.

Description of amendment request: The proposed amendment would change Technical Specification (TS) Section 6.9, ''Administrative Controls—Reporting Requirements,'' to eliminate the requirement to submit startup reports to the Nuclear Regulatory Commission. Under the current provisions of TS Section 6.9, the Davis-Besse Nuclear Power Station would be required to submit a startup report within 90 days.

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

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

Response: No.

The proposed change is administrative in nature. As such, it does not affect any accident initiators and does not affect containment isolation, plant responses to accidents, or radiological effluents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change is administrative in nature. As such, it does not introduce any new or indifferent accident initiators. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previous evaluated.

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

Response: No.

The proposed change is administrative in nature and does not reduce or adversely affect the capabilities of any plant structures, systems, or components to perform their safety functions. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: June 20, 2003.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs) to allow a one-time extension of the interval between integrated leakage rate tests from 10 years to 15 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:

Proposed Power Level Changes

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

Response: No

Probability of Occurrence of an Accident Previously Evaluated—

The proposed change to extend the [integrated leakage rate tests] ILRT interval from 10 to 15 years does not affect any accident initiators or precursors. The containment vessel function is purely mitigative. There is no design basis accident that is initiated by a failure of the containment leakage mitigation function. The extension of the ILRT will not create any adverse interactions with other systems that could result in initiation of a design basis accident. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated—

The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from

10 to 15 years. The increase in risk in terms of person rem per year within 50 miles resulting from design basis accidents was estimated to be of a magnitude that NUREG–1493 indicates is imperceptible. NMC has also analyzed the increase in risk in terms of the frequency of large early releases from accidents. The increase in the large early release frequency resulting from the proposed extension was determined to be within the guidelines published in Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMC has determined that the increase in conditional containment failure probability from reducing the ILRT frequency from 1 test per 10 years to 1 test per 15 years would be small. Continued containment integrity is also assured by the history of successful ILRTs, and that established programs for local leakage rate testing and in-service inspections which are unaffected by the proposed change. Therefore, the consequences of an accident previously analyzed are not significantly increased.

In summary, the probability of occurrence and the consequences of an accident previously evaluated are not significantly increased.

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

Response: No.

The proposed change to extend the ILRT interval from 10 to 15 years does not create any new or different accident initiators or precursors. The length of the ILRT interval does not affect the manner in which any accident begins. The proposed change does not create any new failure modes for the containment and does not affect the interaction between the containment and any other system. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The risk-based margins of safety associated with the containment ILRT are those associated with the estimated person-rem per year, the large early release frequency, and the conditional containment failure probability. NMC has quantified the potential effect of the proposed change on these parameters and determined that the effect is not significant. The non-risk-based margins of safety associated with the containment ILRT are those involved with its structural integrity and leak tightness. The proposed change to extend the ILRT interval from 10 to 15 years does not adversely affect either of these attributes. The proposed change only affects the frequency at which these attributes are verified. Therefore, the proposed change does not involve a significant reduction in margin of safety.

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

Attorney for licensee: John H. O'Neill, Jr., Esq., Shaw Pittman, Potts & Trowbridge, 2300 N. Street, NW, Washington, DC 20037–1128.

NRC Section Chief: L. Raghavan.

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

Date of amendment requests: June 26, 2003.

Description of amendment requests: The proposed license amendment would update the Diablo Canyon Power Plant (DCPP) Final Safety Analysis Report Update to use a revised steam generator voltage-based repair criteria probability of detection method for DCPP Unit 2 Cycle 12 using plant-specific inspection results.

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 use of a revised steam generator (SG) voltage-based repair criteria probability of detection (POD) method, the probability of prior cycle detection (POPCD) method, to determine the beginning of cycle (BOC) indication voltage distribution for the Diablo Canyon Power Plant (DCPP) Unit 2 Cycle 12 operational assessment (OA) does not increase the probability of an accident. Based on industry and plant specific bobbin detection data for outside diameter stress corrosion cracks (ODSCC) within the SG tube support plate (TSP) region, large voltage bobbin indications which individually can challenge structural or leakage integrity can be detected with near 100 percent certainty. Since large voltage ODSCC bobbin indications within the SG TSP can be detected, they will not be left in service, and therefore these indications should not be included in the voltage distribution for the purpose of OAs. POPCD improves the estimate of potentially undetected indications for OAs, but does not directly affect the inspection results. Since large voltage indications are detected, they will not result in an increase in the probability of a steam generator tube rupture (SGTR) accident or an increase in the consequences of a SGTR or main steam line break (MSLB) accident.

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

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

The use of the POPCD method to determine the BOC voltage distribution for the DCPP Unit 2 Cycle 12 OA concerns the SG tubes and can only affect numerical predictions of probabilities for the SGTR accident. Since the SGTR accident is already considered in the Final Safety Analysis Report Update, there [is] no possibility to create a design basis accident that has not been previously evaluated.

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

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

The use of the POPCD method to determine the BOC voltage distribution for the DCPP Unit 2 Cycle 12 OA does not involve a significant reduction in a margin of safety. The applicable margin of safety potentially impacted is the Technical Specification 5.6.10, "Steam Generator Tube Inspection Report," projected end-of-cycle leakage for a MSLB accident and the projected end-of-cycle probability of burst. Based on industry and plant specific bobbin detection data for ODSCC within the SG TSP region, large voltage bobbin indications that can individually challenge structural or leakage integrity can be detected with near 100 percent certainty and will not be left in service. Therefore these indications should not be included in the voltage distribution for the purpose of OAs. Since these large voltage indications are detected, they will not result in a significant increase in the actual end-of-cycle leakage for a MSLB accident or the actual end-of-cycle probability of burst. The POPCD approach to probability of detection considers the potential for missing indications that might challenge structural or leakage integrity by applying the POPCD data from successive inspections. If a large indication was missed in one inspection, it would continue to grow until finally detected in a later inspection.

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: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: July 3, 2003.

Description of amendment request: The proposed amendments would change the Technical Specifications (TSs) 3.8.1 for AC Sources—Operating, to extend, on a one-time basis, the allowable Completion Time for Required Actions associated with one offsite circuit inoperable, from 72 hours to 10 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposal would change the Technical Specifications for AC Sources—Operating, to extend, on a one-time basis, the allowable Completion Times for Required Actions for one offsite circuit inoperable, from 72 hours to 10 days. The proposed change does not involve a significant increase in the probability of an accident previously evaluated because the probability increases are within the guidance provided in Regulatory Guide 1.177.

The consequence[s] of losing offsite power have been evaluated in the FSAR [Final Safety Analysis Report] and the Station Blackout evaluation. Increasing the completion time for one offsite power source from 72 hours to 10 days does not increase the consequences of a LOOP [loss of offsite power] event nor change the evaluation of LOOP events as stated in the FSAR or Station Blackout evaluation.

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

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

Response: No.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed nor will there be changes in methods governing normal plant operation).

Allowing the completion time for ST [startup transformer] No. 10 to increase from 72 hours to 10 days is a one-time change that will allow continued operation of Unit 1 while replacing Startup Transformer Number 10. The accident analyses affected by this extension are the LOOP events that are discussed in the FSAR. The potential for the loss of other plant systems or equipment to mitigate the effects of an accident is not altered.

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

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

Response: No

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

The proposed change allows, on a one-time basis, ST No. 10 to be out of service for 7 days more than is allowed by Technical Specifications. This increase in completion time for ST No. 10 results in a slight decrease in the margin of safety. Implementation of the compensatory measures described in Section 4.0 mitigates the increase in the core damage frequency and large early release frequency during this time, such that the potential impact of extending the completion time is small. Therefore, this one-time exemption will not involve a significant reduction in safety margin.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Richard J. Laufer.

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

Date of application request: June 2, 2003.

Description of amendment request: The amendment would increase the value of the minimum fuel oil required in the storage tank for the emergency diesel generators in Technical Specification (TS) 3.8.3, ''Diesel Fuel Oil, Lube Oil, and Starting Air.'' The licensee stated it has implemented the change in the field. This was done because the proposed new value is higher than the current value in the TSs.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The design of the emergency diesel engine fuel oil storage and transfer system and the function of the onsite standby power sources will be unaffected. The only physical change is to increase the [minimum] volume of fuel oil required to run the emergency diesel generators at their continuous rating for 6 days. This change has already been implemented in the field and is in the conservative direction. The fuel oil storage and transfer system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to this amendment request are maintained.

The probability and consequences of accidents previously evaluated in the FSAR [(Callaway Final Safety Analysis Report)] are not adversely affected because the change to the [minimum] volume of fuel oil required is conservative and is consistent with the safety analysis and licensing basis.

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

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

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of plant operation or change any operating parameters. The proposed change does not induce a new mechanism that would result in a different kind of accident from those previously analyzed. No performance requirements or response time limits will be affected.

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

This amendment does not alter the performance of the emergency diesel engine fuel oil storage and transfer system in [its] support of the onsite standby power sources.

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

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

The proposed change does not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. The minimum volume of fuel oil required for a 6[-]day supply as specified in the TS has already been increased in the conservative direction. The safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis are changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan will continue to be met.

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

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

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

NRC Section Chief: Stephen Dembek

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

Date of application request: June 6, 2003

Description of amendment request: The amendment would modify several surveillance requirements (SRs) in Technical Specifications (TSs) 3.8.1 and 3.8.4 on alternating current and direct current sources, respectively, for plant operation. The revised SRs would have notes deleted or modified to allow the SRs to be performed, or partially performed, in reactor modes that are currently not allowed by the TSs. The current SRs are not allowed to be performed in Modes 1 and 2. Several of the current SRs also cannot be performed in Modes 3 and 4.

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 design of plant equipment is not being modified by the proposed changes. In addition, the DGs [diesel generators] and their associated emergency loads are accident mitigating features. As such, testing of the DGs themselves is not associated with any potential accident-initiating mechanism. Therefore, there will be no significant impact on any accident probabilities by the approval of the requested changes.

The changes include an increase in the online time that a DG under test will be paralleled to the grid (for SRs 3.8.1.10 and 3.8.1.14) or unavailable due to testing (per SR 3.8.1.13). As such, the ability of the tested DG to respond to a design basis accident [(DBA)] could be adversely impacted by the proposed changes. However, the impacts are not considered significant based, in part, on the ability of the remaining DG to mitigate a DBA or provide safe shutdown. With regard to SR 3.8.1.10 and SR 3.8.1.14, experience shows that testing per these SRs typically does not perturb the electrical distribution system. In addition, operating experience and qualitative evaluation of the probability of the DG or bus loads being adversely affected concurrent with or due to a significant grid disturbance, while the DG is being tested, support the conclusion that the proposed changes do not involve any significant increase in the likelihood of a safety-related bus blackout or damage to plant loads.

The SR changes that are consistent with TSTF [Technical Specification Task Force]—283 have been approved by the NRC for submittal by licensees. The on-line tests allowed by the TSTF are only to be performed for the purpose of establishing OPERABILITY. Performance of these SRs during restricted MODES will require an assessment to assure plant safety is maintained or enhanced.

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

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

The capability to synchronize a DG to the offsite source (via the associated plant bus) and test the DG in such a configuration is a design feature of the DGs, including the test mode override in response to a safety injection signal. Paralleling the DG for longer periods of time during plant operation may slightly increase the probability of incurring an adverse effect from the offsite source, but this increase in probability is judged to be still quite small and such a possibility is not a new or previously unrecognized consideration.

The proposed changes would not require any new or different accidents to be postulated since no changes are being made to the plant that would introduce any new accident causal mechanisms. This license amendment request does not impact any plant systems that are potential accident initiators; nor does it have any significantly adverse impact on any accident mitigating systems.

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

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

The proposed changes do not involve a significant reduction in the margin of safety. The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design [safety] functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes do not directly affect these barriers, nor do they involve any significantly adverse impact on the DGs which serve to support these barriers in the event of an accident concurrent with a loss of offsite power. The proposed changes to the testing requirements for the plant DGs do not affect the OPERABILITY requirements for the DGs, as verification of such OPERABILITY will continue to be performed as required (except during different allowed MODES [of operation]). The changes have an insignificant impact on DG availability, as continued verification of OPERABILITY supports the capability of the DGs to perform their required [safety] function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Only one DG is to be tested at a time, so that the remaining DG will be available to safely shut down the plant if required. Consequently, performance of the fission product barriers will not be impacted by implementation of the proposed amendment.

In addition, the proposed changes involve no changes to [safety] setpoints or limits established or assumed by the accident analys[e]s. On this and the above basis, no safety margins will be impacted.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: June 27, 2003.

Description of amendment request: The amendment would revise the technical specifications (TSs) in two parts. It would: (1) Revise the definition of dose equivalent radioiodine 131 (I–131) by adding the phrase ''or those derived from the data provided in International Commission on Radiological Protection Publication 30 (ICRP 30), 'Limits for Intakes of Radionuclides by Workers,' 1979'' to the current definition, and (2) increase the maximum allowed closure time of each main feedwater isolation valve (MFIV) from 5 seconds to 15 seconds in Surveillance Requirement 3.7.3.1. A plant modification would replace the electro-hydraulic MFIV actuators with system-medium actuators to improve MFIV reliability and reduce maintenance requirements. The MFIV stroke time would be increased. A plant modification would also replace swing check valves in each auxiliary feedwater (AFW) motor-driven pump discharge line with an automatic recirculation control (ARC) check valve to reduce the potential for vibration and increase AFW flow margin.

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.

MFIV Actuator Replacement and Increased MFIV Stroke Time

[* * *], the increase in MFIV stroke time does not adversely impact the NSSS [nuclear steam system supplier] design transients evaluated for the Callaway Plant. The increase in MFIV stroke time will result in a slightly longer normal post trip cool down. Although the plant post trip cool down is expected to be slightly longer for the increased MFIV stroke time, the plant response does not significantly deviate from its current evaluated response following a normal reactor trip.

Evaluations assessing the impact of the change in MFIV actuators and the increase in MFIV stroke time on LOCA [loss-of-coolant accident] mass and energy releases; main steamline break mass and energy releases; LOCA and LOCA[-]related transients; non-LOCA transients; LOCA hydraulic forces[;] and steam releases used for radiological consequence calculations were also performed. The increase in isolation time and change in MFIV actuators either do not provide an adverse impact or have no impact. Except for the SGTR [steam generator tube rupture)] with overfill accident, the results presented in the FSAR [Callaway Final Safety Analysis Report] remain valid. The increase in MFIV stroke time was evaluated for impact on the SGTR with overfill accident. [* * *], the results from the re-analysis of the SGTR with overfill accident confirm that there is no significant increase in the probability or consequences of an accident previously evaluated.

The replacement of the existing electro-hydraulic MFIV actuators with system-medium actuators and the increase in MFIV stroke time from 5 seconds to 15 seconds will not result in a significant increase in the probability or consequences of an accident previously evaluated.

MDAFP [Motor Driven Auxiliary Feedwater Pump] ARC Valve and Increased Maximum AFW Flow

The replacement of existing MDAFP discharge check valves with the ARC valves results in increased maximum AFW flow to the steam generators [(SGs)]. In many accident scenarios the increase in AFW flow to the SGs is beneficial to mitigation of the event. The evaluations [* * *] demonstrate that in those accident scenarios where maximum AFW flow is limiting, except for the SGTR with overfill accident, the increase in AFW flow remains bounded by FSAR analyses. The increase in maximum AFW flow was evaluated for impact on the SGTR with overfill accident. [* * *], the results from the re-analysis of the SGTR with overfill accident confirm that there is no significant increase in the probability or consequences of an accident previously evaluated. The AFW system is not the initiator of any accident and there is no possibility of a significant increase in the probability of an accident or malfunction previously evaluated.

Use of the ARC valve is an enhancement and the associated increase in the maximum AFW flow will not result in a significant increase in the probability or consequences of an accident previously evaluated.

Use of Revised Methods in Re-Analysis of SGTR With Overfill

The re[-]analysis of the design basis accident for SGTR with overfill does not significantly increase the probability or consequences of an accident previously evaluated. The re-analysis of an accident is not an initiator [of an accident]. The SGTR accident is classified as an ANS [American Nuclear Society] Condition IV Event, Limiting Faults, and is only postulated and not expected to occur. The re[-]analysis activity being evaluated does not change the ANS classification for this design basis event. The re-analysis does provide dose consequences that are minimal increases to the doses in the Analysis of Record.

However, the doses remain well below regulatory limits. In support of this methodology the proposed TS definition for DOSE EQUIVALENT I–131 will allow the use of ICRP 30 based DCFs [dose conversion factors]. Section 4.1.2 and [* * *] Appendix E of Regulatory Guide 1.195 find acceptable and recommend the [proposed] method revisions.

In summary, using the proposed revised methods for the re-analysis of the SGTR with overfill does not result in a significant increase in the probability or consequences of an accident previously evaluated.

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

MFIV Actuator Replacement and Increased MFIV Stroke Time

The change in MFIV actuators and associated increase in MFIV stroke time will not prevent the main feedwater or auxiliary feedwater systems from performing their safety functions. The proposed increase will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the increase. Although the modification does alter the design of the MFIV actuators, it does not prevent the main feedwater or AFW systems from performing their safety functions.

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

MDAFP ARC Valve and Increased Maximum AFW Flow

The new MDAFP ARC valve and associated increase in the maximum AFW flow [* * *] will not prevent the AFW system from performing its safety function. The proposed increase in AFW system flow margin will not effect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the increase in AFW system flow margin. Although the modification alters the design of the MDAFP discharge check valves, it does not prevent the AFW system from performing its safety functions.

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

Use of Revised Methods in Re-Analysis of SGTR With Overfill

The revision to the Technical Specifications to allow the use of ICRP 30[-]based DCFs is based on methodologies found acceptable to the NRC and recommended for use as described in Section 4.1.2 of Regulatory Guide 1.195. The re[-]analysis of the design basis accident for SGTR with overfill and the use of recommended analysis methods acceptable to the NRC does not introduce the possibility of a new accident. Accident re-analysis is not an initiator of any accident and no new failure modes are introduced. In summary, there is no increase in the possibility of an accident of a different type.

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

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

MFIV Actuator Replacement and Increased MFIV Stroke Time

The replacement of the MFIV actuator and the associated increase in the MFIV stroke time does not affect the manner in which safety limits or limiting safety system settings are determined, nor will there be any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no significant impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss[-]of[-]coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan will continue to be met.

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

MDAFP ARC Valve and Increased Maximum AFW Flow

The use of the MDAFP ARC valve and the associated increase in AFW system flow margin does not affect the manner in which safety limits or limiting safety system settings are determined nor will there be any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no significant impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss[-]of[-]coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

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

Use of Revised Methods in Re-Analysis of SGTR With Overfill

Use of revised methods in the re-analysis for the SGTR with overfill accident does not affect the manner in which safety limits or limiting safety system settings are determined nor will there be any adverse effect on those plant systems necessary to assure the accomplishment of protection functions. There is no significant impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor (F-delta-H), loss[-]of[-]coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met. The re-analysis of the SGTR with overfill confirms that both the thermal-hydraulic and radiological consequences are within the regulatory requirements and does not result in a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: June 27, 2003.

Description of amendment request: The amendment would revise the Technical Specifications (TSs) to (1) extend the allowed outage time (AOT) or required action completion time (CT) for an inoperable diesel generator (DG) by adding the phrase ''*OR* 108 hours once per cycle for each DG'' to the completion time for Required Action B.4 in TS 3.8.1, ''AC Sources—Operating,'' and (2) delete the second CT given in certain required actions in TS 3.6.6, ''Containment Spray and Cooling Systems''; TS 3.7.5, ''Auxiliary Feedwater (AFW) System''; TS 3.8.1; and TS 3.8.9, ''Distribution System—Operating,'' of the TSs. The second part would also delete Example 1.3–3, delete text referring to this example, and re-number the remaining examples in TS 1.3, ''Completion Times.''

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.

DG AOT/CT Extension

The proposed change to extend the DG AOT/CT from 72 hours to 108 hours for planned, on-line maintenance does not affect the design of the DGs, the operational characteristics or function of the DGs, the interfaces between the DGs and other plant systems, or the reliability of the DGs. The DGs mitigate the consequences of previously evaluated accidents including loss[-]of[-]offsite power, but as such are not themselves initiators of any previously evaluated accidents. DG allowed outage time is thus not associated with any initiating condition for accidents previously evaluated. The consequences of an accident are independent of the time the DGs are out of service as long as adequate DG availability is assured. The proposed changes will not result in a significant decrease in DG availability, so assumptions regarding DG availability are not impacted. Since the DGs will continue to be capable of performing their accident mitigation function as assumed in the accident analysis, the consequences of accidents previously analyzed are unchanged with respect to the proposed changes.

In addition, to fully evaluate the effect of the proposed DG completion time extension, probablistic risk assessment methods and a deterministic analysis were utilized. The results of the analyses show no significant increase in core damage frequency or large early release frequency.

Elimination of Second Completion Times

Similar to the above change, the changes to eliminate the ''second'' Completion Times from the affected Technical Specifications [(*i.e.*, the specific TS sections being changed)] do not affect the design, operational characteristics, or intended functions of the equipment addressed by those Technical Specifications. With no direct effects on that equipment (or any other plant equipment or features), allowed equipment outage times are not associated with any initiating condition for any accident previously evaluated, and therefore would not affect the probability of such accidents. Further, eliminating these Completion Times is not expected to have an adverse effect on the availability of the applicable systems or components because equipment availability performance criteria required for conformance to the Maintenance Rule impose an equivalent or acceptable level of control and management of equipment availability regardless of such Completion Times. As noted above, the consequences of evaluated accidents are independent of mitigating equipment allowed outage times as long as adequate availability of the equipment is ensured. Since elimination of the second Completion Times has no significant impact on equipment availability (in light of continued, required conformance to the Maintenance Rule), the consequences of accidents previously evaluated are unchanged.

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.

None of the proposed changes, *i.e.*, neither the DG AOT extension nor the elimination of [the] second Completion Times, involve a change in the design, configuration, or operational characteristics of the plant. No physical alteration of the plant is involved, as no new or different type of equipment is to be installed. The changes do not alter any assumptions made in the safety analyses, and no alteration in the procedures which ensure that the plant remains within analyzed limits is being proposed. As such, no new failure modes or mechanisms that could cause a new or different kind of accident from any previously evaluated are being introduced.

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

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

The proposed DG AOT extension and elimination of second Completion Times do not alter the manner in which safety limits or limiting safety system settings are determined. The safety analysis acceptance criteria are not impacted by [these] change[s], and the proposed changes will not permit plant operation in a configuration [that is] outside the design basis.

Further, with regard to plant risk, the risk assessment performed for the DG AOT extension determined that the quantifiable increase in plant risk is acceptably small. Likewise, for the elimination of [the] second Completion Times, it may be assumed that this change also involves little or no increase in risk on the basis that required, continued compliance with the Maintenance Rule provides adequate controls for maintaining equipment availability regardless of the second Completion Times [proposed to be eliminated].

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: John O'Neill, Esq., Shaw, Pittman, Potts &

Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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

Date of application request: June 27, 2003.

Description of amendment request: The licensee is proposing to amend the operating license for the Callaway Plant to allow plant modifications in order to facilitate maintenance on the replacement steam generators (SGs) to be installed in Refueling Outage (RO) 14 (Fall 2005). The proposed modifications (1) replace the existing sludge lance platforms with new platforms to provide a larger platform area around each SG, and (2) cut a permanent access opening through the secondary shield wall to improve access to the sludge lance platforms. They are to be done in RO 13 (Spring 2004). Dynamic effects associated with large reactor coolant system (RCS) branch line ruptures are to be excluded using a proposed leak-before-break (LBB) methodology. The amendment would authorize changes to the Callaway licensing basis to be added to the Callaway Final Safety Analysis Report (FSAR). There are no proposed changes 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 proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall protection system performance will remain within the bounds of the previously performed accident analyses. The design of the protection systems will be unaffected. The reactor protection system and engineered safety feature actuation system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

Neither the currently intact ''C'' SG cubicle secondary shield wall, nor the proposed configuration that provides a permanent access opening, create accident initiation mechanisms that would increase the probability of an accident. There will be no change to normal plant operating parameters or accident mitigation performance.

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

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

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

There are no changes in the method by which any safety-related plant system performs its safety function. This amendment will not affect the normal method of power operation or change any operating parameters. No performance requirements will be affected, but SG maintenance access will be greatly improved.

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

Presence of a permanent access opening in the ''C'' loop SG secondary shield wall does not, of itself, create the possibility of a new accident since the secondary shield walls are not used for missile protection and the high-energy line breaks (greater than 10-inches in diameter) that would generate missiles will be removed from the structural design basis after NRC's review and acceptance of the LBB topical reports.

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

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

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

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss[-]of[-]coolant accident peak cladding temperature (LOCA PCT), or peak local power density. The LBB margins discussed in NUREG–1061 Volume 3 are satisfied. The radiological dose consequence acceptance criteria listed in the [NRC] Standard Review Plan will continue to be met. The secondary shield walls are not fission product barriers. They provide radiation shielding to maintain occupational exposure ALARA [as low as is reasonably achievable] and provide structural support to primary coolant SSCs [structures, systems, and components].

The proposed amendment does not eliminate any surveillances or alter the Frequency of surveillances required by the Technical Specifications. The nominal Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) trip setpoints (TS Bases Tables B 3.3.1–1 and B 3.3.2–1), RTS and ESFAS allowable values (TS Tables 3.3.1–1 and 3.3.2–1), and the safety analysis limits assumed in the transient and accident analyses (FSAR Table 15.0–4) are unchanged. None of the acceptance criteria for any accident analysis is changed.

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

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

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

NRC Section Chief: Stephen Dembek.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and 2, Louisa County, Virginia

Date of amendment request: March 28, 2002, as supplemented by letters dated May 13, June 19, and November 15, 2002, and May 6, May 9, May 27, and June 11 (2 letters), 2003. This notice supersedes the notice that was published on May 14, 2002 (67 FR 34496).

Description of amendment request: The proposed amendments would permit Virginia Electric and Power Company to replace the existing Westinghouse fuel with Framatome ANP Advanced Mark-BW fuel at North Anna Power Station, Units 1 and 2. This submittal was accompanied by requested exemptions from the requirements of 10 CFR 50.44 and 10 CFR 50.46. These exemptions will be processed separately.

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

1. The probability of occurrence or the consequences of an accident previously evaluated is not significantly increased.

The proposed methodology has been generically reviewed and approved for use by the NRC for determining core operating limits prior to its use by Dominion. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits developed in accordance with the new methodologies will be bounded by any limitations in the NRC safety evaluation report (SER) for the new methodologies. Application of the topical reports associated with the new methodologies will demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed changes do not involve physical changes to any plant structure, system, or component. Therefore, the probability of occurrence of any accident previously evaluated is not significantly increased.

The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed changes do not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose resulting from an accident. The proposed changes do not affect setpoints that initiate protective or mitigative actions. The proposed changes ensure that plant structures, systems, and components are maintained consistent with the safety analysis and licensing basis. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created.

The proposed changes do not involve any physical alteration of plant systems, structures, or components, other than allowing for fuel design in accordance with NRC-approved methodologies. The proposed methodologies continue to meet applicable criteria for LBLOCA [large-break loss-of-coolant accident] and SBLOCA [small-break loss-of-coolant accident] analyses. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result, no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized in response to plant transients changed. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced.

The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within safety analyses. The proposed changes in the methodologies used in the LBLOCA and SBLOCA analyses do not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed changes in the analysis methodologies comply with the requirements of 10 CFR 50.46 paragraph (a)(1)(i) (*i.e.*, not exceeding a peak cladding temperature of 2200°F for [SB] LOCA and a high probability that peak cladding temperature will remain below 2200°F for [LB] LOCA). Therefore, the margin of safety as defined in the Bases to the North Anna Units 1 and 2 Technical Specifications is not significantly reduced.

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: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

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, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov*.

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

Date of application for amendments: April 15, 2003.

Brief description of amendments: The amendments revise Sections 2.2, ''SL [Safety Limits] Violations,'' for reporting such violations to positions in the plant organization; 5.2.1, ''Onsite and Offsite Organization,'' for the position responsible for overall safe plant operation; and 5.5.1, ''Offsite Dose Calculation Manual (ODCM),'' to replace the positions of Vice President, Nuclear Production, and Director, Site Chemistry, with other positions in the plant organization. Also, there would be the format change of adding the title of Section 2.2 near the top of TS page 2.0–2.

Date of issuance: June 26, 2003.

Effective date: June 26, 2003, and shall be implemented within 30 days of the date of issuance.

Amendment Nos.: Unit 1–146, Unit 2–146, Unit 3–146.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 27, 2003 (68 FR 28845).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 2003.

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: August 26, 2002, as supplemented by letter dated June 18, 2003.

Brief description of amendments: The amendments revise the Technical Specifications regarding the Diesel Fuel Oil Testing Program.

Date of issuance: July 10, 2003.

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

Amendment Nos.: 206 & 200.

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

*Date of initial notice in **Federal Register:*** October 29, 2002 (67 FR 66008).

The supplement dated June 18, 2003, provided clarifying information that did not change the scope of the August 26, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 10, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 12, 2002, as supplemented by letters dated March 27 and April 23, 2003.

Brief description of amendments: The amendments revise the Technical Specifications (TS) regarding the reactor vessel pressure-temperature limit curves and revise the low-temperature overpressure protection limits. The licensee also requested that a change be made to TS Table 3.3.2–1, Footnote (c) to correct what was claimed to be an editorial error. This request was not supported by sufficient information and, accordingly, is denied.

Date of issuance: July 3, 2003.

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

Amendment Nos.: 214 & 195.

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

*Date of initial notice in **Federal Register:*** January 7, 2003 (68 FR 801).

The supplement dated March 27 and April 23, 2003, provided clarifying information that did not change the scope of the December 12, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 3, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 26, 2002, as supplemented by letter dated June 18, 2003.

Brief description of amendments: The amendments revise the Technical Specifications regarding the Diesel Fuel Oil Testing Program.

Date of issuance: July 10, 2003.

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

Amendment Nos.: 215 & 195.

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

*Date of initial notice in **Federal Register:*** October 29, 2002 (67 FR 66008).

The supplement dated August 26, 2002, provided clarifying information that did not change the scope of the June 18, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 10, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 19, 2003.

Brief description of amendment: The amendment deletes Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' and License Condition 2(C)(33)(c) from Facility Operating License NPF–29, thereby eliminating the requirement to have and maintain the post-accident sampling system at Grand Gulf Nuclear Station, Unit 1. The amendment also addresses related changes to TS 5.5.2, ''Primary Coolant Sources Outside Containment.''

Date of issuance: June 30, 2003.

Effective date: As of the date of issuance and shall be implemented within 120 days of issuance.

Amendment No: 158.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications and deletes License Condition 2(C)(33)(c).

*Date of initial notice in **Federal Register:*** May 13, 2003 (68 FR 25652).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 30, 2003.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: January 13, 2003, supplements dated February 27, March 6, March 14, April 30, June 9, and June 30, 2003.

Brief description of amendment: The proposed amendment would revise the Kewaunee Nuclear Power Plant Facility Operating License and Technical Specifications to increase the licensed rated power by 1.4 percent from 1650 megawatts thermal to 1673 megawatts thermal using measurement uncertainty recapture.

Date of issuance: July 8, 2003.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 168.

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

*Date of initial notice in **Federal Register:*** February 4, 2003 (68 FR 5679).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 8, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 29, 2003.

Brief description of amendments: The amendments revise Salem, Unit No. 1, Technical Specifications (TSs) Section 3/4.7.6, and Salem, Unit No. 2, TSs 3/4.2.2, 3/4.7.6, and Table 3.3–6. These changes are administrative and editorial in nature, and correct errors made during the implementation of previously-approved TS changes.

Date of issuance: June 26, 2003.

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

Amendment Nos.: 258 and 239.

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

*Date of initial notice in **Federal Register:*** April 15, 2003 (68 FR 18284).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 26, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 11th day of July 2003.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 03–18084 Filed 7–21–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Fire Dynamics Tools (FDTs)—Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, Availability of NUREG

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability.

SUMMARY: The Nuclear Regulatory Commission is announcing the completion and availability of Draft NUREG–1805, ''Fire Dynamics Tools (FDTs)—Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program,'' dated June 30, 2003.

ADDRESSES: Draft NUREG–1805 is available for inspection and copying for a fee at the NRC Public Document Room, 11555 Rockville Pike, Rockville, Maryland. As of July 8, 2003, you may also electronically access NUREG-series publications and other NRC records at NRC's Public Electronic reading Room at *www.nrc.gov/reading-rm.html.*

A free single copy of Draft NUREG–1805, to the extent of supply, may be requested by writing to Office of the Chief Information Officer, Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Printing and Graphics Branch, Washington, DC 20555–000; facsimile: 301–415–2289; e-mail: *DISTRIBUTION@nrc.gov.*

Some publications in NUREG-series that are posted at NRC's Web site address *www.nrc.gov/NRC/NUREGS/indexnum.html* are updated regularly and may differ from the last printed version.

FOR FURTHER INFORMATION CONTACT: Naeem Iqbal or Mark H. Salley, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Telephone: 301–415–3346 or 301–415–2840.

SUPPLEMENTARY INFORMATION: The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (NRR), Division of Systems Safety and Analysis (DSSA), Plant Systems Branch (SPLB), Fire Protection Engineering and Special Projects Section has developed quantitative methods, known as ''Fire Dynamics Tools (FDTs),'' to assist regional fire protection inspectors in performing fire hazard analysis (FHA). These methods have been implemented in spreadsheets and taught at the NRC's quarterly regional inspector workshops. The goal of the training is to assist inspectors in calculating the quantitative aspects of a postulated fire and its effects on safe nuclear power plant (NPP) operation. FDTs were developed using state-of-the-art fire dynamics equations and correlations that were pre-programmed and locked into Microsoft Excel® spreadsheets. These FDTs will enable the inspector to perform quick, easy, first-order calculations for the potential fire scenarios using today's state-of-the-art principles of fire dynamics. Each FDTs spreadsheet also contains a list of the physical and thermal properties of the materials commonly encountered in NPPs.

The FDTs are intended to assist fire protection inspectors in performing risk-informed evaluations of credible fires that may cause critical damage to essential safe-shutdown equipment. This is the process required by the new reactor oversight process (ROP) in the NRC's inspection manual. In the new ROP, the NRC is moving toward a more risk-informed, objective, predictable, understandable, and focused regulatory process. Key features of the new program are a risk-informed regulatory framework, risk-informed inspections, a significance determination process (SDP) to evaluate inspection findings, performance indicators, a streamlined assessment process, and more clearly defined actions that the NRC will take for plants based on their performance.

This NUREG addresses the technical bases for FDTs, which were derived from the principles developed primarily in the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering, National Fire Protection Association (NFPA) Fire Protection Handbook, and other fire science literature. The subject matter of this NUREG covers many aspects of fire dynamics and contains descriptions of the most important fire processes. A significant number of examples, reference tables, illustrations, and conceptual drawings are presented in this NUREG to expand the inspector's appreciation in visualizing and retaining the material and understanding calculation methods.

The content of the FDTs encompasses fire as a physical phenomenon. As such, the inspector needs a working knowledge of algebra to effectively use the formulae presented in this NUREG and FDTs. Acquired technical knowledge or course background in the sciences will also prove helpful. The information contained in this NUREG is similar to, but includes less theory and detail than, an undergraduate-level university curriculum for fire protection engineering students.

The goal of this NUREG is to develop a common body of knowledge of commercial NPP fire protection and fire science to enable the inspector to acquire the understanding, skills, and abilities necessary to effectively apply principles of fire dynamics to analyze the potential effects of a fire in an NPP. The FDTs will advance the FHA process from a primarily qualitative approach to a more quantitative approach. The development of this NUREG, the FDTs, and the quarterly inspector workshops conducted in 2001–2002 are the NRC's first steps in achieving that goal.

Fire is a complex subject and transfer of its concepts to useful pursuits is a challenge. We hope that this NUREG and the FDTs can make a difference in the NRC's fire protection inspection program, specifically risk-informed fire protection initiatives such as the SDP and risk-informed inspection of associated circuits.

Dated at Rockville, Maryland, this 23 day of June, 2003.

For the Nuclear Regulatory Commission.

John N. Hannon,

Chief, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation.

[FR Doc. 03–18543 Filed 7–21–03; 8:45 am]

BILLING CODE 7590–01–P

Overseas Private Investment Corporation

Agency Report Form Under OMB Review

AGENCY: Overseas Private Investment Corporation (OPIC).

ACTION: Request for comments.

SUMMARY: Under the provisions of the Paperwork Reduction Act (44 U.S.C. Chapter 35), agencies are required to publish a Notice in the **Federal Register** notifying the public that the Agency has