collection request; Abstract: Instructors will be asked to complete a questionnaire before the course begins. In most cases, it will be specified in the instructor's contract that they complete the questionnaire. This survey instrument contains six questions, most requiring only a simple fill-in-the blank response. Data and information from this questionnaire will establish a baseline of the instructor's expectations and intentions to be used in measuring changes at the end of the course. Affected Entities: Entities potentially affected by this action are instructors who lead training sessions sponsored by the U.S. Institute. Burden Statement: It is estimated that the annual national public burden and associated costs will be approximately 33.3 hours and $1,200, respectively. These values were calculated assuming that on average: (a) instructors require 10 minutes to complete the questionnaire; and (b) each year there are 200 instructors who work on training sessions sponsored by the U.S. Institute. Cost burden estimates assume: (a) there are no capital or start-up costs for respondents, and (b) respondents' time is valued at $36 hr.

(17) Training—Instructor Questionnaire, at the conclusion; New collection request; Abstract: When the course concludes, instructors will be asked to complete a questionnaire. In most cases, it will be specified in their contract that they complete this questionnaire. The survey instrument contains five questions, most requiring only a simple fill-in-the blank response. Data and information from this questionnaire will help establish a contextual baseline for evaluating survey data from the training participants. As well, this instrument is also intended to generate useful feedback on ways to improve the U.S. Institute's training projects. Affected Entities: Entities potentially affected by this action are instructors who lead training sessions sponsored by the U.S. Institute. Burden Statement: It is estimated that the annual national public burden and associated costs will be approximately 33.3 hours and $1,200, respectively. These values were calculated assuming that on average: (a) instructors require 10 minutes to complete the questionnaire; and (b) each year there are 200 instructors who work on training sessions sponsored by the U.S. Institute. Cost burden estimates assume: (a) there are no capital or start-up costs for respondents, and (b) respondents' time is valued at $36 hr.

Meeting Facilitation

U.S. Institute staff and contractors facilitate and provide leadership for many public meetings, ranging from small group meetings to large public convenings of several hundred attendees. In order to maximize the probability that such meeting objectives will be accomplished, the meeting participants must both understand the objectives for the meeting, and perceive that the meeting was managed in a fair and efficient manner. This requires that the right facilitator run the meeting, and the right people attend the meeting.

(18) Meeting Facilitation—Meeting Attendees Questionnaire, at the conclusion of the process; New collection request; Abstract: Attendees at public meetings run by U.S. Institute staff or contractors will be asked to complete a voluntary questionnaire at the conclusion of the meeting. The questionnaire used in this case contains nine questions, two-thirds requiring only a simple fill-in-the blank response. Information from this questionnaire will help evaluate the effectiveness of individual facilitators and particular meeting process designs. Affected Entities: Entities potentially affected by this action are individuals who participate in these public meetings. Burden Statement: It is estimated that the annual national public burden and associated costs will be approximately 833.3 hours and $22,500, respectively. These values were calculated assuming that on average: (a) meeting attendees require 10 minutes to complete the questionnaire; (b) the U.S. Institute conducts 100 public meetings each year; and (c) 50 people attend the average meeting. Cost burden estimates assume: (a) there are no capital or start-up costs for respondents; and (b) respondents' time is valued at $27 hr.

Dated: December 18, 2001.

Christopher L. Helms,

Executive Director, Morris K. Udall Foundation.

[FR Doc. 01–31587 Filed 12–21–01; 8:45 am]

BILLING CODE 6820–FN–P

NATIONAL SCIENCE FOUNDATION

DOE/NSF Nuclear Science Advisory Committee; Notice of Meeting

In accordance with the Federal Advisory Committee Act (Pub. L. 92–463, as amended), the National Science Foundation announces the following meeting.

Name: DOE/NSF Nuclear Science Advisory Committee (1176).

Date and Time: Monday, Jan. 14, 2002 8 a.m.–6 p.m.; Tuesday, Jan. 15, 2002; 8 a.m.–6 p.m.

Place: Rm 585–II 4201 Wilson Blvd., Arlington, VA 22230.

Contact Person: Dr. Bradley D. Keister, Program Director for Nuclear Physics, National Science Foundation, 4201 Wilson Blvd., Arlington, VA 22230. Telephone (703) 292–7380.

Purpose of Meeting: To provide advice and recommendations concerning the scientific programs of the NSF and DOE in the area of basic nuclear physics research.

Dated: December 19, 2001.

Susanne Bolton,

Committee Management Officer.

[FR Doc. 01–31640 Filed 12–21–01; 8:45 am]

BILLING CODE 7555–01–M

NUCLEAR REGULATORY COMMISSION

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

(**Note:** The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.)

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from December 3, 2001 through December 14, 2001. The last biweekly notice was published on December 12, 2001 (66 FR 64284).

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 NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 25, 2002, 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 NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically 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/NRC/ADAMS/index.html. 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 Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.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, 304–415–4737 or by email to *pdr@nrc.gov.*

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

Date of amendment request: April 17, 2001.

Description of amendment request: The proposed amendment would make editorial and administrative corrections to Technical Specifications (TS) Section 3.3, ''Instrumentation'', and eliminate minor discrepancies between TS Section 3.3 and other plant licensing basis documents.

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

Does the Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The proposed changes involve correction of editorial or administrative errors made during the conversion of the Clinton Power Station (CPS) Technical Specifications (TS) to the improved TS (ITS). These proposed changes are based upon current design and licensing basis requirements. The proposed changes involve correction or reformatting of the TS and do not involve any physical changes to plant systems, including those that mitigate the consequences of accidents or the manner in which these plant systems are operated. As such, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Does the Change Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

The proposed changes involve correcting errors or reformatting existing TS requirements that do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. These changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the Change Involve a Significant Reduction in a Margin of Safety?

The proposed changes involve correcting editorial or administrative errors introduced during the conversion of the CPS TS to the ITS. The change to the Allowable Value for the Control Room Ventilation System air intake radiation monitors setpoint in TS Table 3.3.7.1–1 is consistent with the supporting analyses for the trip setpoint value that was previously contained in the TS. The changes involve reformatting or correction of errors, and therefore will not reduce any margin of safety because there is no effect on any safety analysis assumptions. These proposed changes maintain requirements within the safety analyses and licensing basis. Therefore, these 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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: May 21, 2001.

Description of amendment request: The proposed amendment would revise the actions required if the refueling equipment interlocks become inoperable. The proposed changes are consistent with the changes submitted to the Nuclear Regulatory Commission by the Technical Specifications Task Force, Issue number 225, Revision 1.

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

Does the Change Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated?

The proposed addition of alternate actions in the event that the refueling equipment interlocks are determined to be in operable ensures that the safety function provided by the interlocks are enforced. This is accomplished through manually inserting a rod block to prevent the inadvertent withdrawal of a control rod when fuel is being moved over the core region.

The refueling equipment interlocks are credited in the Control Rod Removal Error During Refueling—Fuel Insertion with Control Rod Withdrawn as described in Updated Safety Analysis (USAR Section 15.4.1.1.2.2). The manual insertion of a control rod withdrawal block provides equivalent protection for the conditional rod block provided by the refueling equipment interlocks.

The proposed change to the surveillance frequency does not change the means in which the refueling equipment operates. A review of surveillance history was performed for the past two refueling outages. In the last seven performances of the refueling equipment interlocks operability test, the interlocks have operated successfully with no corrective maintenance or corrective action necessary. Therefore, since the proposed changes do not result in any physical changes to the facility, or involve any modifications to plant systems or design parameters or conditions that contribute to the initiation of any accidents previously evaluated, the proposed changes do not increase the probability of any accident previously evaluated.

Since the proposed changes maintain the same level of protection provided by the refueling equipment interlocks, the conclusion of the accident scenario remain valid. The probability of a criticality event during refueling remains such that no radioactive material would be released. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

Does the Change Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated?

The proposed changes do not involve a change to the plant design or operation. Inserting a manual rod block is not considered an abnormal operation. The change to the SR [surveillance requirement] frequency does not increase the probability of a malfunction of the refueling equipment interlocks, since the interlocks are considered reliable and their function can be verified with each fuel move. As a result, the proposed changes do not affect any of the parameters or conditions that could contribute to the initiation of any accidents. No new accident modes or equipment failure modes are created by these changes.

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

Does the Change Involve a Significant Reduction in a Margin of Safety?

The major challenge to the margin of safety would be a criticality event that would cause a potential failure of the fuel cladding. The proposed addition of alternative actions in the event that the refueling equipment interlocks are determined to be inoperable ensure that equivalent protection is in place during fuel loading movements. Given this equivalent protection, a criticality event is not credible. In addition, the increase in the SR frequency for performing the channel functional test of the refueling equipment interlocks does not impact the ability of the interlocks to perform their function, thereby maintaining the refueling interlocks function.

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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 4300 Windfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

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: December 7, 2001.

Description of amendment request: The proposed amendments would allow implementation of 10 CFR Part 50, Appendix J, Option B, which governs performance-based containment leakage testing requirements, for Type B and C testing. In addition, the licensee also proposes to (a) modify Technical Specification (TS) 3.6.3 to delete the requirement for conducting soap bubble tests of welded penetrations during Type A tests which are not individually Type B or Type C testable, and (b) to modify TS 3.6.3 to delete a separate requirement for leak testing containment purge lower and upper compartment and instrument room valves with resilient seals. These valves will be covered by the overall Containment Leakage Rate Testing Program. Associated changes to the Bases are also proposed.

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

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if 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.

First Standard

The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Implementation of these changes will provide continued assurance that specified parameters associated with containment integrity will remain within acceptance limits as delineated in 10 CFR 50, Appendix J, Option B. The changes are consistent with current safety analyses. Although some of the proposed changes represent minor relaxation to existing TS requirements, they are consistent with the requirements specified by Option B of 10 CFR 50, Appendix J. The systems affecting containment integrity related to this proposed amendment request are not assumed in any safety analyses to initiate any accident sequence. Therefore, the probability of any accident previously evaluated is not increased by this proposed amendment. The proposed changes maintain an equivalent level of reliability and availability for all affected systems. In addition, maintaining leakage within analyzed limits assumed in accident analyses does not adversely affect either onsite or offsite dose consequences. Therefore, the proposed amendment does not increase the consequences of any accident previously evaluated.

Second Standard

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No changes are being proposed which will introduce any physical changes to the existing plant design. The proposed changes are consistent with the current safety analyses. Some of the changes may involve revision in the testing of components; however, these are in accordance with the McGuire's current safety analyses and provide for appropriate testing or surveillance that is consistent with 10 CFR 50, Appendix J, Option B. The proposed changes will not introduce new failure mechanisms beyond those already considered in the current safety analyses. No new modes of operation are introduced by the proposed changes. The proposed changes maintain, at minimum, the present level of operability of any system that affects containment integrity.

Third Standard

The proposed amendment will not involve a significant reduction in a margin of safety. The provisions specified in Option B of 10 CFR 50, Appendix J allow changes to Type B and Type C test intervals based upon the performance of past leak rate tests. 10 CFR 50, Appendix J, Option B allows longer intervals between leakage tests based on performance trends, but does not relax the leakage acceptance criteria. Changing test intervals from those currently provided in the TS to those provided in 10 CFR 50, Appendix J, Option B does not increase any risks above and beyond those that the NRC has deemed acceptable for the performance based option. In addition, there are risk reduction benefits associated with reduction in component cycling, stress, and wear associated with increased test intervals. The proposed changes provide continued assurance of leakage integrity of containment without adversely affecting the public health and safety and will not significantly reduce existing safety margins. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to McGuire.

Based upon the preceding discussion, Duke Energy has concluded that the proposed amendment does not involve a significant hazards consideration.

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

Attorney for licensee: Mr. Albert Carr, Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: November 15, 2001.

Description of amendment request: Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station (GGNS) Operating License be amended to revise the GGNS Technical Specification Surveillance Requirements (SRs) pertaining to testing of the standby emergency diesel generators (DGs) to allow DG testing during reactor operation. The proposed change would remove the restriction associated with these SRs that prohibits conducting the required testing of the DGs during reactor operating Modes 1, 2, or 3.

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 any accident previously evaluated.

The DGs and their associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. As such, the ability of the DGs to respond to a design basis accident will not be adversely impacted by these proposed changes. The capability of the DGs to supply power in a timely manner will not be compromised by permitting performance of DG testing during periods of power operation. Additionally, limiting testing to only one DG at a time ensures that design basis requirements for backup power is met, should a fault occur on the tested DG. Therefore, there would be no significant impact on any accident consequences.

Based on the above, the proposed change to permit certain DG surveillance tests to be performed during plant operation will have no effect on accident probabilities or consequences.

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

No new accident causal mechanisms would be created as a result of NRC [Nuclear Regulatory Commission] approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration with the exception of the plant mode in which the testing is conducted. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

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

(3) The proposed change does not involve a significant reduction in the 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 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).

Continued verification of operability supports the capability of the DGs to perform their required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted. Therefore, implementation of the proposed changes would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 29, 2001.

Description of amendment request: The proposed amendments would revise technical specification (TS) 3.9.3, ''Refueling Operations—Decay Time,'' by reducing the amount of time that the reactor must be subcritical before the licensee is allowed to move irradiated fuel assemblies in the reactor pressure vessel from 150 hours to 100 hours. The amendment also makes various editorial, format and administrative changes.

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

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

No. The proposed change does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed change does not alter the manner in which heavy loads are controlled at BVPS. The proposed change does not result in changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. Also, the proposed change does not impact the design of plant systems such that previously analyzed SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the NRC approved fuel handling accident (FHA) analyses for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. The BVPS radiological analyses demonstrates that should a FHA occur within the containment or the fuel building that involves irradiated fuel with at least 100 hours of decay, the projected offsite doses for this event will be well within the applicable regulatory limits.

Limiting Condition for Operation (LCO) 3.9.3, ''Refueling Operations—Decay Time,'' will continue to ensure that irradiated fuel is not moved in the reactor pressure vessel until at least 100 hours after shutdown which is consistent with the FHA radiological analysis. This LCO will continue to ensure that key assumptions used in the radiological safety analysis are met. The previously analyzed SSCs are unaffected by the proposed change and continue to provide assurance that they are capable of performing their intended design function in mitigating the effects of design basis accidents (DBAs). As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

The proposed administrative, editorial, and format changes do not affect the probability or consequences of any accident.

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

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

No. The proposed amendment does not affect a previously evaluated accident; e.g., FHA. The proposed amendment takes credit for the normal decay of irradiated fuel and the existing radiological analyses for FHAs.

The proposed change does not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the technical specification. The proposed change does not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor does it alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change.

The proposed change does not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed change has no effect on previously evaluated accidents.

The proposed administrative, editorial, and format changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

No. The proposed revision of the decay time from 150 hours to 100 hours is consistent with the assumptions used in the NRC approved FHA accident analyses for BVPS Unit Nos. 1 and 2 and thus does not involve a significant reduction in a margin of safety.

The proposed amendment does not alter the manner in which fuel assemblies are handled or core alterations are performed. The proposed amendment does not alter the manner in which heavy loads are controlled at BVPS.

The proposed changes to the TS requirements will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA. The proposed administrative, editorial, and format changes do not affect plant safety.

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

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: October 31, 2001.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) by relocating the pressure temperature Limit Curves and Low Temperature Overpressure Protection (LTOP) and by creating a Pressure-Temperature Limits Report in accordance with Generic Letter 96–03 (GL–96–03), ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.''

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. The proposed changes are a relocation of the Reactor Coolant System (RCS) pressure/temperature (P/T) limits, overpressure protection system (OPPS) setpoint, and the enable temperature from the Technical Specifications to the proposed Pressure and Temperature Limits Report (PTLR). The PTLR is created in accordance with the guidance provided by Generic Letter (GL) 96–03 and is consistent with the content of NUREG–1431. The RCS P/T limits, OPPS setpoint, and enable temperature will continue to meet the requirements of 10 CFR 50, Appendix G, and will be generated in accordance with the NRC approved methodology described in WCAP–14040–NP–A, Rev. 2 with the exceptions noted in Technical Specification Section 6.9.6.

Since the proposed changes are administrative in nature and do not involve any change to any values being relocated, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. As stated above, the proposed changes to relocate the RCS P/T limits, OPPS setpoint, and the enable temperature from the Technical Specifications to the PTLR are administrative changes. The proposed changes do not result in a physical change to the plant or add any new or different operating requirements on plant systems, structures, or components.

Therefore, the proposed changes do not result in a significant increase in the possibility of a new or different accident from any previously evaluated.

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

No. The margin of safety is not affected by the creation of the proposed PTLR. Operation of the plant in accordance with the limits specified in the PTLR will continue to meet the requirements of 10 CFR 50, Appendix G, with the identified exceptions, and will assure that a margin of safety is not significantly decreased as the result of the proposed changes.

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

NRC Section Chief: Lakshminaras Raghavan (Acting).

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

Date of amendment request: October 9, 2001.

Description of amendment request: The proposed amendment changes affected Technical Specifications (TS) 3/4.3.2.2, ''Instrumentation—Steam and Feedwater Rupture Control System Instrumentation,'' including Table 3.3–11, ''Steam and Feedwater Rupture Control System Instrumentation,'' Table 3.3–12, ''Steam and Feedwater Rupture Control System Instrumentation Trip Setpoints,'' and Table 4.3–11 ''Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements.'' Related administrative changes are proposed to TS 3/4.3.2.3, ''Instrumentation—Anticipatory Reactor Trip System Instrumentation,'' Table 3.3–17, ''Anticipatory Reactor Trip System Instrumentation,'' and TS 3/4.3.3.1, ''Instrumentation—Monitoring Instrumentation—Radiation Monitoring Instrumentation,'' Table 3.3–6, ''Radiation Monitoring Instrumentation.'' Related changes to associated TS Bases 3/4.3.1 and 3/4.3.2, ''Reactor Protection System and Safety System Instrumentation,'' are also proposed.

The main purpose for this license amendment request is to decrease the channel functional test frequency from monthly to quarterly for the Steam and Feedwater Rupture Control System (SFRCS) Instrumentation Channels.

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. These changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated because the proposed changes do not change any accident initiator, initiating condition, or assumption.

The proposed changes would revise Technical Specification (TS) Table 3.3–11, ''Steam and Feedwater Rupture Control System Instrumentation,'' and Table 4.3–11 ''Steam and Feedwater Rupture Control System Instrumentation Surveillance Requirements,'' to identify the Steam and Feedwater Rupture Control System (SFRCS) output logic as a separate Functional Unit. In addition, the proposed changes would revise TS Table 3.3–12, ''Steam and Feedwater

Rupture Control System Instrumentation Trip Setpoints,'' to remove the ''Trip Setpoint'' values and also modify the ''Allowable Values'' entry for Functional Unit 3, ''Steam Generator Feedwater Differential Pressure—High,'' consistent with updated calculations and current setpoint methodology, and revise the applicability of TS Allowable Values for other SFRCS Functional Units in this table. The proposed changes would also revise TS Table 4.3–11 to change the Channel Functional Test surveillance requirements for the SFRCS instrument channels from monthly to quarterly, consistent with current methodology. The proposed changes would also make related administrative changes to TS Limiting Condition for Operation (LCO) 3.3.2.2, TS Table 3.3–17, ''Anticipatory Reactor Trip System Instrumentation,'' TS Table 3.3–6, ''Radiation Monitoring Instrumentation,'' and the associated TS Bases.

These proposed changes do not involve a significant change to plant design or operation.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed changes do not invalidate assumptions used in evaluating the radiological consequences of an accident, do not alter the source term or containment isolation, and do not provide a new radiation release path or alter radiological consequences.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed changes do not introduce a new or different accident initiator or introduce a new or different equipment failure mode or mechanism.

3. Not involve a significant reduction in a margin of safety as defined in the basis for any Technical Specification. The SFRCS instrumentation setpoint analyses will continue to adequately preserve the margin of safety. In addition, there are no new or significant changes to the initial conditions contributing to accident severity or consequences. Therefore, there are no significant reductions in a margin of safety.

Conclusion:

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: October 12, 2001.

Description of amendment request: The proposed amendment would change the Operating License (OL) paragraph 2.C(1), Maximum Power Level; OL paragraph 2.C(3)(d), Additional Conditions; Technical Specification (TS) 1.3, Definitions—Rated Thermal Power; TS 2.1.1, Safety Limits—Reactor Core, and associated Bases; TS 2.2.1, Limiting Safety System Settings—Reactor Protection System Setpoints, and associated Bases; TS 3/4.1.1.3, Reactivity Control Systems—Moderator Temperature Coefficient; TS 3/4.2.5, Power Distribution Limits—DNB Parameters; TS 3/4.4.9.1, Reactor Coolant System—Pressure/Temperature Limits, and associated Bases; and TS 6.9.1.7, Core Operating Limits Report. The purpose of this license amendment application would make the necessary revisions to the Davis-Besse Nuclear Power Station (DBNPS) TS to reflect an increase in the authorized rated thermal power from 2772 MWt to 2817 MWt (approximately 1.63 percent), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System instrumentation to improve the accuracy of the feedwater mass flow input to the plant power calorimetric measurement.

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:

1a. Not involve a significant increase in the probability of an accident previously evaluated based on the comprehensive analytical efforts that were performed to demonstrate the acceptability of the proposed power uprate changes. The proposed changes include: revision of the maximum power level limit stated in Operating License (OL) paragraph 2.C(1) and Technical Specification (TS) Section 1.3, increasing the allowable power level from 2772 MWt to 2817 MWt; revision of the reactor core safety limits specified in TS Section 2.1.1; revision of the Reactor Protection System (RPS) high flux and Reactor Coolant System (RCS) pressure-temperature setpoints provided in TS Section 2.2.1; revision of the RCS pressure-temperature limits in TS Section 3/4.4.9.1, and a related change to OL paragraph 2.C(3)(d); and revision of administrative controls associated with the Core Operating Limits Report, as described in TS Section 6.9.1.7. In addition, related changes to the TS Bases associated with these TS Sections are proposed. An evaluation has been performed that identified the systems and components that could be affected by these proposed changes. The evaluation determined that these systems and components will function as designed and that performance requirements remain acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pumps, steam generators and pressurizer) will continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components leading to an accident.

The Leak-Before-Break analysis conclusions remain valid and the breaks previously exempted from structural consideration remain unchanged.

All of the Nuclear Steam Supply System (NSSS) systems will continue to perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses, which credit the flow, do not require any modification. The components continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the NSSS/Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The main steam safety valves will provide adequate relief capacity to maintain the main steam system within design limits.

The current loss of coolant accident (LOCA) hydraulic forcing functions remain bounding.

The reduction in power measurement uncertainty through the use of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system, allows for certain safety analyses to continue to be used, without modification, at the 2827 MWt power level (102% of 2772 MWt). Other safety analyses performed at a nominal power level of 2772 MWt have been either re-performed or re-evaluated at the 2817 MWt power level, and continue to meet their applicable acceptance criteria. Some existing safety analyses had been previously performed at a power level greater than 2827 MWt, and thus continue to bound the 2817 MWt power level.

The proposed changes to the RCS pressure-temperature limit curves impose a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50 Appendix G, ''Fracture Toughness Requirements,'' will continue to be met following the proposed power uprate. The design basis events that were protected against by these limits have not changed, therefore, the probability of an accident previously evaluated is not increased.

In addition to the changes related to the proposed power uprate, unrelated changes are proposed to revise the moderator temperature coefficient requirements listed in TS Section 3.1.1.3, and to revise requirements relating to the Departure from Nucleate Boiling (DNB) parameters listed in TS Section 3.2.5. These proposed changes are conservative changes and clarifications that do not involve any physical change to systems or components, nor do they alter the typical manner in which the systems or components are operated. Therefore, these changes will not result in a significant increase in the probability of an accident.

1b. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed power uprate changes do not alter any assumptions previously made in the radiological consequence evaluations, nor affect mitigation of the radiological consequences of an accident previously evaluated.

The accident radiation dose evaluation was performed at 2827 MWt and is bounding when operating at the proposed 2817 MWt using the LEFM CheckPlus™ flow instrumentation.

The proposed changes unrelated to the power uprate also do not alter any assumption previously made in the radiological consequence evaluations, nor do they affect mitigation of the radiological consequences of an accident previously evaluated. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

2. Not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed power uprate changes as well as the proposed changes unrelated to the power uprate. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system.

3. Not involve a significant reduction in a margin of safety because extensive analyses of the primary fission product barriers, conducted in support of the proposed power uprate, have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of at the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC) or that are in compliance with applicable regulatory review guidance and standards. The proposed changes unrelated to the power uprate do not involve a significant reduction in a margin of safety because they do not involve the potential for a significant increase in a failure rate of any system or component, and existing system and component redundancy is not affected. Also, these changes do not involve any new or significant changes to the initial conditions contributing to accident severity or consequences.

Conclusion:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 26, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) to delete Section 3/4.2.6, ''Inservice Inspection and Testing,'' and its associated bases, revise Section 4.2.7, ''Reactor Coolant System Isolation Valves,'' and its associated bases, create a new Section 6.17, ''Inservice Testing Program,'' and delete several reporting requirements in Section 6.9.3, ''Special Reports.'' These changes will improve the TSs, making it consistent with current NRC guidance and the improved Standard Technical Specifications for General Electric (GE) Boiling Water Reactor (BWR)/4 and BWR/6 plants (NUREG–1433 and NUREG–1434, respectively). Most of these changes would also render the TSs to be similar to the Nine Mile Point Nuclear Station, Unit No. 2 TSs, which is based on NUREG–1433 and NUREG–1434.

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

1. The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment deletes duplicative and unnecessary inservice inspection (ISI) and inservice testing (IST) requirements from the Technical Specifications; clarifies remaining IST requirements; revises a requirement to perform quarterly testing of the reactor coolant isolation valves to conform to the periodic testing requirements of the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code (ASME Code); and deletes unnecessary reporting requirements relating to routine ISI, primary containment leakage testing, and secondary containment leakage testing. These changes do not reduce the plant's existing ISI/IST commitments based on 10CFR50.55a, Section XI of the ASME Code, and Generic Letter 88–01. These changes also do not involve hardware changes, changes in plant setpoints, or changes in plant safety parameters.

Based on the above, the operation of Nine Mile Point Unit 1 (NMP1) in accordance with the proposed amendment, will not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve any physical modifications to the plant nor alter equipment configuration, setpoints, or safety parameters. The ISI/IST related changes are consistent with current NRC guidance and industry standards and will continue to ensure acceptable equipment operability and availability.

Based on the above, the operation of NMP1 in accordance with the proposed amendment cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The proposed changes do not affect any of the plant's fission product barriers or safety/operational limits. The ISI/IST related changes will continue to ensure acceptable equipment operability and availability.

Based on the above, the operation of NMP1 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: November 20, 2001.

Description of amendment request: The licensee proposed to amend the Technical Specifications (TSs) regarding the safety limit minimum critical power ratio (SLMCPR) to reflect the results of cycle-specific calculations performed for the next fuel cycle (i.e., Cycle 9), using NRC-approved methodology for determining SLMCPR values. The proposed amendment would also editorially revise references to topical reports which document the approved methodology.

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

1. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the revised Safety Limit Minimum Critical Power Ratio (SLMCPR) values for Nine Mile Point Unit 2 (NMP2) Cycle 9 for incorporation into the Technical Specifications (TS) and their use to determine cycle-specific thermal limits has been performed using the NRC-approved methods and procedures in [Topical Report] NEDE–24011–P–A, ''General Electric Standard Application for Reactor Fuel'' (GESTAR II). The analysis methodology incorporates cycle-specific parameters and reduced power distribution uncertainties in the determination of the SLMCPR values. These calculations do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the Minimum Critical Power Ratio Safety Limit is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and the probability of fuel damage is not increased. The deletion of listed documents that are already incorporated by reference into GESTAR II is administrative only. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The new SLMCPR values for the NMP2 Cycle 9 core reload have been calculated in accordance with the methods and procedures described in GESTAR II. These methods have been reviewed and approved by the NRC. The deletion of listed documents that are already incorporated by reference into GESTAR II is administrative only. The changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or transients result from these changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The margin of safety as defined in the TS bases will remain the same. The new, cycle-specific SLMCPR values are calculated using NRC-approved methods and procedures that are in accordance with the current fuel design and licensing criteria. The SLMCPR values remain high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limits are not violated, thereby preserving the fuel cladding integrity. The deletion of listed documents that are already incorporated by reference into GESTAR II is administrative only. Therefore, the proposed TS 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 requested amendment involves no significant hazards consideration.

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: November 19, 2001.

Description of amendment request: A change is proposed to Technical Specification (TS) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to the specification: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The 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. 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 following NSHC determination in its application dated November 7, 2001.

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

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

The 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.

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

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.

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

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 [Limiting Condition for Operation] 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.

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

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

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

NRC Section Chief: William D. Reckley, Acting.

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

Date of amendment request: February 2, 2001, supplemented August 31, 2001.

Description of amendment request: The proposed amendments would revise the technical specifications (TSs) to clarify the plant conditions under which various specifications are applicable. The licensee stated in its amendment request that a literal reading of the current technical specifications wording may result in situations where a routine plant shutdown would seem to be prohibited by TSs and, thereby, require entry into TS 3.0.C. This amendment request also makes several administrative changes to the TSs, including revising references to the Chief Nuclear Corporate Officer, capitalizing defined terms, and updating references to previously relocated TS paragraphs and correcting the List of Figures. The licensee's supplement to the amendment request, dated August 31, 2001, proposed a correction of a typographical error in TS Table 3.5–2B, Action 33.

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 operation of the facility with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes are administrative in nature and clarify existing specifications without reducing or altering the requirements imposed by existing specifications. The proposed changes do not significantly affect any system that is a contributor to initiating events for previously evaluated accidents. Neither do the changes significantly affect any system that is used to mitigate any previously evaluated accidents. Therefore, the proposed changes do not involve any significant increase in the probability or consequence of an accident previously evaluated.

2. Does operation of the facility with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes are administrative in nature and clarify existing specifications without reducing or altering the requirements imposed by existing specifications. The proposed changes do not alter the design, function, or operation of any plant component and do not install any new or different equipment, therefore a possibility of a new or different kind of accident from those previously analyzed has not be[en] created.

3. Does operation of the facility with the proposed amendment involve a significant reduction in a margin of safety?

The proposed changes are administrative in nature and clarify existing specifications without reducing or altering the requirements imposed by existing specifications. Thus, the proposed change[s] do not involve a significant reduction in the margin of safety associated with the safety limits inherent in either the principle barriers to a radiation release (fuel cladding, RCS [reactor coolant system] boundary, and reactor containment), or the maintenance of critical safety functions (subcriticality, core cooling, ultimate heat sink, RCS inventory, RCS boundary integrity, and containment integrity).

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

NRC Acting Section Chief: William D. Reckley, Acting.

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

Date of amendment request: November 21, 2001.

Description of amendment request: The proposed amendment will revise Technical Specifications 2.15(5) and 2.15(6) to identify: (1) all indication and control functions required for the alternate (remote) shutdown panels, (2) panel locations of the functions, and (3) the number of operable channels required.

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

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

The proposed changes to Technical Specifications Sections 2.15(5) and 2.15(6) identify functions, instruments, and controls along with their location and the number of required channels. New Technical Specifications Section 2.15(5) addresses the regulatory requirements for equipment required for Alternative and Dedicated Shutdown Capability per 10 CFR part 50, Appendix R. It will ensure that proper Limiting Conditions for Operation are entered for equipment or functional inoperability. There are no physical alterations being made to the Alternate Shutdown Panels and the Auxiliary Feedwater Panel or related systems. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes will not result in any physical alterations to the Alternate Shutdown Panels or the Auxiliary Feedwater Panel, or any plant configuration, systems, equipment, or operational characteristics. There will be no changes in operating modes, or safety limits, or instrument limits. With the proposed changes in place, Technical Specifications retain requirements for the Alternate Shutdown Panels and the Auxiliary Feedwater Panel. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed changes clarify the regulatory requirements for the Alternative and Dedicated Shutdown Capability as defined by 10 CFR Part 50, Appendix R. The proposed changes will not alter any physical or operational characteristics of the Alternate Shutdown Panels and the Auxiliary Feedwater Panel and their associated systems and equipment. Therefore, the proposed changes do not involve a reduction in a margin of safety.

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: November 21, 2001.

Description of amendment request: The proposed amendment will add three topical report references to Technical Specification 5.9.5, ''Core Operating Limit Reports.''

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

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

The proposed amendment incorporates three additional Framatome ANP topical reports for conducting core reload analyses. Since the intent of the amendment request is to add references to NRC-approved reload analysis methods, 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.

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

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

As required by Technical Specification 5.9.5, the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The proposed change incorporates methodologies applicable for use with fuel supplied by Framatome ANP that have been approved by the NRC as documented by Safety Evaluation Reports (References 10.1, 10.2, and 10.3 [of the November 21, 2001, amendment request]). Because Technical Specification 5.9.5 also requires that the core operating limits shall be determined and requires that all applicable limits of the safety analysis are met, 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: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Pacific Gas and Electric Company, Docket No. 50–133, Humboldt Bay Power Plant, Unit 3, Humboldt County, California

Date of amendment request: December 28, 2000, as supplemented by letters dated March 29 and October 31, 2001.

Description of amendment request: The proposed amendment would convert the Humboldt Bay Power Plant Unit 3 Current Technical Specifications to a set of Permanently Defueled Technical Specifications with a more standardized format and content based on a revision to 10 CFR 50.36 (Technical Specifications) and technical specifications approved for other permanently shutdown nuclear power plants (Millstone Unit 1 and Trojan Nuclear Plant).

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

The conversion of the Humboldt Bay Power Plant (HBPP) Current Technical Specifications (CTS) to Permanently Defueled Technical Specifications (PDTS) involves the following four types of dispositions:

A Administrative reformatting and rewording
D Item deleted from the Technical Specifications (TS)
LG Relocating items from CTS to the Defueled Safety Analysis Report (DSAR), PDTS, or other Licensee-Controlled Document
N Addition of new requirements of new sections to the PDTS

Administrative Reformatting and Rewording

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

The proposed change involves reformatting and editorially rewording of the CTS. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this 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 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 not impose any different operational requirements and any administrative additions are non-operational in nature and have not been identified and justified. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

Items Deleted from the Technical Specifications that are Duplicative in Nature to Other Regulatory Requirements

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

The proposed change involves deleting information from the CTS. The information being deleted is still required to be performed and is being performed by the licensee because the information is contained in regulatory requirements contained in the Code of Federal Regulations. Therefore, this 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 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 not impose any different operational requirements. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. The requirements being deleted from the CTS are still required to be met and are being met by the licensee because these requirements exist in the Code of Federal Regulations. As such, no question of safety is involved.

Items Deleted from the Technical Specifications That Have No Application in the Proposed HBPP PDTS

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

The proposed change involves deleting information from the CTS. The deletion process involves no technical changes to the CTS. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this 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 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 not impose any different operational requirements. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

Relocating Information from CTS to the DSAR, PDTS Bases or Other Licensee-Controlled Documents

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

The proposed change relocates requirements and descriptive information from the CTS to the PDTS Bases, DSAR, or other licensee-controlled documents. The PDTS Bases, DSAR, or other licensee-controlled documents containing the relocated requirements and information will be maintained using provisions of 10 CFR 50.59 or other appropriate regulatory controls. Since any future changes to the PDTS Bases, DSAR, or other licensee-controlled documents will be evaluated per the requirements of 10 CFR 50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this 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 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 not impose any different requirements and adequate control of the information will be maintained. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements and information to be relocated from the CTS to the PDTS Bases, DSAR, or other licensee-controlled documents are not being revised; they are being relocated verbatim. Since any future changes to these requirements in the PDTS Bases, DSAR, or other licensee-controlled documents will be evaluated per the requirements of 10 CFR 50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore this change does not involve a significant reduction in a margin of safety.

Addition of New Requirements or New Sections to the PDTS

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

The proposed change involves the addition of requirements or sections to the proposed PDTS. Each addition either provides equivalent or potentially more restrictive controls than previously provided. The additional requirements or controls do not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this 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 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 not impose any different operational requirements and any addition is non-operational in nature and has been identified and justified. Thus, this 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 will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change provides the equivalent or more restrictive requirements on the surveillance and control of TS parameters. As such, no question of safety is involved.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: August 22, 2001.

Description of amendment request: Approve reactor core power uprate, and revise the Technical Specifications to reflect the power uprate.

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:

STPNOC [South Texas Project Nuclear Operating Company] has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of amendment,'' as discussed below.

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

The comprehensive analytical efforts performed to support the proposed uprate conditions include a review and evaluation of all components and systems (including interface systems and control systems) that could be affected by this change. The revised power uprate value was input to applicable safety analyses. The proposed change is not an initiator of any design-basis accident. All of the Nuclear Steam Supply System or Balance of Plant interface systems will continue to perform their intended design functions and meet all performance requirements. The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms, loop piping and supports, reactor coolant pump, steam generator, and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Therefore, there is no increase in the probability of a structural failure of these components.

The auxiliary systems and components continue to comply with applicable structural limits and will continue to perform their intended design functions. Therefore, there is no increase in the probability of a structural failure of these components. The steam generator safety valves will provide adequate relief capacity to maintain the steam generators within design limits. The steam dump system will still relieve 40 percent of the maximum full-load steam flow.

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

The applicable analyses have been evaluated with respect to the increase in core power associated with this change. All applicable radiological acceptance criteria continue to be met. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change neither causes the initiation of any accident nor creates any new limiting single failures. All of the affected systems and components continue to perform their intended design functions. The proposed change has no adverse effects on any safety-related system or component and does not challenge the performance or integrity of any safety-related system.

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 WRB–2M DNB methodology is used to demonstrate that core thermal-hydraulic limits are maintained without any significant reduction in margin of safety for the uprated power level of 3853 MWt (1.4-percent uprate) assuming core designs composed of Robust Fuel Assemblies. The WRB–1 DNB correlation demonstrates that there is no significant reduction in the margin of safety for core designs composed of standard or Vantage 5 Hybrid (V5H) fuel types. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that either have been reviewed and approved by the Nuclear Regulatory Commission or are in compliance with all applicable regulatory review guidance and standards.

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

Based on the above, STPNOC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 23, 2001.

Brief description of amendments: The proposed amendment would revise Technical Specification (TS) 5.5.9, ''Steam Generator Tube Surveillance Program,'' to permit tube sleeving repair techniques, developed by Westinghouse Electric Company (Westinghouse) and referred to as ''Westinghouse Leak Tight Sleeves.''

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.

The Westinghouse Leak Tight Sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applicable design criteria for the sleeves conforms to the stress limits and margins of safety of Section III of the ASME code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Draft Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

Evaluation of the repaired steam generator tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor coolant system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at CPSES [Comanche Peak Steam Electric Station]. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The installation of the proposed sleeves is controlled via the sleeving vendor's proprietary processes and equipment. The Westinghouse process has been in use since 1984 and has been implemented more than 24 times for the installation of over 4,200 sleeves. The CPSES steam generator design was reviewed and found to be compatible with the installation processes and equipment.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeved tube is bounded by the current steam generator tube rupture (SGTR) analysis described in the CPSES FSAR [Final Safety Analysis Report]. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis, depending on the break location, and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feed line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the CPSES safety analysis. The proposed reduction of the steam generator primary to secondary operational leakage limit provides added assurance that leaking flaws will not propagate to burst prior to commencement of plant shutdown.

In conclusion, based on the discussion above, these changes will not significantly increase 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.

The Westinghouse Leak Tight Sleeves are designed using the applicable ASME Code as guidance; therefore, they meet the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeves. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. As discussed above, the reduced primary to secondary leakage limit is considered a conservative change in the plant limiting conditions for operation. Therefore, TXU Electric concludes that this proposed change does 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.

The repair of degraded steam generator tubes with Westinghouse Leak Tight Sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Code used in the original steam generator design. The portions of the installed sleeve assembly that represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation. Use of the previously identified design criteria and design verification testing assures that the margin of safety is not significantly different from the original steam generator tubes. The proposed sleeve inspection requirements are more stringent than existing requirements for inspection of the steam generator tubes, and the reduction in the operational limit for primary to secondary leakage through the steam generator tubes is more conservative than current requirements. Therefore, TXU Electric concludes that the proposed change does not involve a significant reduction in a margin of safety.

EPRI [Electric Power Research Institute] qualified eddy current techniques will be used for the detection of tube degradation in 3/4 inch welded sleeved tubes. Alternate inspection techniques, may be used as they become available, as long as it can be demonstrated that the technique used provides the same degree or greater degree of inspection rigor.

The effect of sleeving on the design transients and accident analyses were reviewed and found to remain valid up to the level of steam generator tube plugging consistent with the minimum reactor flow rate as specified in Technical Specification 3.4.1. Continued compliance with the RCS [Reactor Coolant System] flow limits of Technical Specification 3.4.1 is assured through precision flow measurements.

Because all relevant safety analyses were reviewed and found to remain valid, and because the appropriate design margins are maintained through compliance with the relevant ASME Code requirements, it is concluded that 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: November 20, 2001.

Description of amendment request: The proposed amendment would: (1) Move Table 4.7.2, ''Primary Containment Isolation Valves'' and references to the Table from the Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TSs) to the Technical Requirements Manual; (2) change Surveillance Requirement 4.7.B.1.b to reflect that the Standby Gas Treatment system (SBGT) duct heater needs to meet relative humidity design basis; (3) add section 3.7.E, ''Reactor Building Automatic Ventilation System Isolation Valves,'' to the Table of Contents; (4) remove wording in 3.5.A.4.a and b referencing a one-time 30-day Limiting Condition for Operation; and (5) make administrative changes to Sections 5.3 and 6.4.

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

The proposed changes consist of removal of the primary containment isolation valve component list from the VY TS, revision of the SBGT inlet heater surveillance minimum power rating and other administrative changes. The probability of occurrence of a previously evaluated accident is not increased because neither containment isolation nor the SBGT heater are accident initiators, and the proposed changes do not impact any accident initiating conditions. The consequences of an accident previously evaluated are not increased because the proposed changes do not impact the ability of containment to restrict, or SBGT to filter, the release of any fission product radioactivity to the environment. The proposed changes to remove the primary containment isolation valve component list from TS, relocate the information to a licensee controlled document, and to change the SBGT inlet heater power input surveillance requirement, will have no significant impact on any safety related structures, systems or components. The TS requirements for the primary containment isolation valves and SBGT operability and surveillance will not be changed. Additionally, the administrative changes do not affect any system operation or function.

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

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

The proposed changes do not involve any physical alteration of plant equipment and do not change the method by which any safety-related system performs its function. No new or different types of equipment will be installed. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

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

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

The proposed administrative changes, the removal of the primary containment isolation valve component list from TS and the change to the SBGT inlet heater power input surveillance requirement, do not alter the TS requirements for containment integrity, containment isolation, SBGT operability, or adversely affect their capability. The changes will not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed changes do no impact design margins of the primary containment isolation system, SBGT or any other system to perform their safety functions. The essential safety functions of providing primary containment integrity and providing filtration of airborne radioactive releases, are maintained. There is no physical or operational change being made which would alter the sequence of events, plant response, or margins in existing safety analyses. The proposed changes result in no impact on analyzed accident event precursors or effects.

These proposed changes do not alter the physical design of the plant. There is no change in methods of operation. The proposed changes do not alter the means by which primary containment isolation capability is maintained and SBGT is operated.

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

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

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

NRC Section Chief: James W. Clifford.

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, 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/NRC/ADAMS/index.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.*

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: May 30, 2001, as supplemented on November 6, 2001.

Brief description of amendment: Connecticut Yankee Atomic Power Company (the licensee) requested changes to the Technical Specifications (TSs) for the Haddam Neck Plant. The changes to Sections 5 and 6 of the TSs correct terminology, clarify the specifications for consistency with established programs and Standard TSs, and reflect current plant conditions. The changes also reflect the licensee's current organization titles. For information only, the licensee also included proposed changes to the TS Bases for spent fuel pool water level and cooling. The NRC staff did not review the proposed changes to the TS Bases.

Date of issuance: December 4, 2001.

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

Amendment No.: 196.

Facility Operating License No. DPR–61: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44164).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 4, 2001.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut Inc., et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut Date of application for amendment: August 9, 2001

Brief Description of amendment: The amendments modify the Millstone Nuclear Power Station, Unit Nos. 2 and 3 Technical Specifications to clarify the licensed operator qualification standards.

Date of issuance: December 5, 2001.

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

Amendment No.: 258 and 199.

Facility Operating License Nos. NPD–69 and NPF–49: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 17, 2001 (66 FR 52798).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 5, 2001.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: April 23, 2001.

Brief description of amendment: The amendment approves a change to Technical Specification (TS) 3.8.1.1, ''Electrical Power System—A.C. Sources.'' The change removes Surveillance Requirement 4.8.1.1.2.c.1 regarding Emergency Diesel Generator inspection at least once per 18 months during shutdown condition.

Date of issuance: December 7, 2001.

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

Amendment No.: 259.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31705).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 7, 2001.

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: June 13, 2000, as supplemented August 30 and September 10, 2001.

Brief description of amendments: The amendments revised the Facility Operating License of each unit to (1) delete license conditions that have been fulfilled; and (2) make other corrections and editorial changes.

Date of issuance: December 5, 2001.

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

Amendment Nos.: 200 and 181.

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

*Date of initial notice in **Federal Register:*** November 1, 2000 (65 FR 65341).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 5, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: June 15, 2001.

Brief description of amendments: Eliminate the Technical Specifications (TS) requirement that the Automatic Depressurization System (ADS) designated Safety/Relief Valves (S/RVs) open during the manual actuation of the ADS and rewords the Surveillance Requirement (SR) 3.5.1.8 frequency to require the testing of all required ADS manual actuation solenoids during the performance of SR 3.5.1.8 in place of testing on a staggered basis.

Date of issuance: December 13, 2001.

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

Amendment Nos.: 151 and 137.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 8, 2001 (66 FR 41618).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 13, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 13, 2001.

Brief description of amendments: These amendments delete Technical Specifications (TS) Section 6.8.4, which required a Post-Accident monitoring program, for Beaver Valley Power Station, Unit Nos. 1 and 2, and thereby eliminate the requirements to have and maintain the post-accident sampling system (PASS) for those units.

Date of Issuance: December 6, 2001.

Effective date: Upon issuance and shall be implemented within 180 days.

Amendment Nos.: 245, 123.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 19, 2001 (66 FR 48286).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 6, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 17, 2001, as supplemented by letter dated September 5, 2001.

Brief description of amendments: The amendments revised Technical Specification (TS) 3/4.9.3, ''Decay Time,'' to allow the start of core offload at 100 hours after reactor subcriticality between September 15 and June 15, when the lake temperature is assumed to be not higher than 77.8°F, and 148 hours after reactor subcriticality between June 16 and September 14, when the lake temperature is assumed to be not higher than 85°F. TS 3/4.9.3 currently prohibits fuel movement in the reactor pressure vessel until the reactor has been subcritical for at least 168 hours. The 168-hour decay time was placed in the CNP TS with Amendment Nos. 169 and 152 to DPR–58 and DPR–74, respectively, on January 14, 1993.

Date of issuance: November 30, 2001.

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

Amendment Nos.: 260 and 243.

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44174)

The supplemental letter 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 amendments is contained in a Safety Evaluation dated November 30, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 30, 2001, as supplemented October 10 and November 16, 2001.

Brief description of amendment: The amendment revises the Technical Specification safety limit minimum critical power ratio for two recirculation pump operation for Cycle 21.

Date of issuance: December 6, 2001.

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

Amendment No.: 125.

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

Date of initial notice in ***Federal Register:*** October 3, 2001 (66 FR 50470)

The October 10 and November 16, 2001, supplements provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 6, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 7, 2001, as supplemented by letters dated October 17 and November 2, 2001.

Brief description of amendment: The requested changes replaced the current accident source term used in the design basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to 10 CFR 50.67, ''Accident Source Term.'' OPPD requested a full implementation of the AST. Changes were also made to the Ft. Calhoun Technical Specifications to make them consistent with the revised associated accident analysis.

Date of issuance: December 5, 2001.

Effective date: December 5, 2001, to be implemented within 60 days from the date of issuance.

Amendment No.: 201.

Facility Operating License No. DPR–40: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 2, 2001 (66 FR 22031).

The October 17 and November 2, 2001, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 5, 2001.

No significant hazards consideration comments received: No.

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

Date of amendments request: May 3, 2001.

Brief Description of amendments: The amendments relocate cycle-specific reactor coolant system parameter limits from the Technical Specifications (TS) and associated Bases, to the Core Operating Limits Report. The amendments also, add a reference to the Refueling Boron Concentration to TS 5.6.5 to correct an omission.

Date of issuance: December 4, 2001.

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

Amendment Nos.: 151 and 143.

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

Date of initial notice in ***Federal Register:*** October 31, 2001 (66 FR 55024).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 4, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 12, 2001.

Brief description of amendments: The amendments consist of deleting Surveillance Requirement 4.4.6.2.2.e of South Texas Project Technical Specifications Section 3/4.4.6.2.

Date of issuance: December 11, 2001.

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

Amendment Nos.: Unit 1—134; Unit 2—123.

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

Date of initial notice in ***Federal Register:*** June 12, 2001 (66 FR 31715).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 11, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of application for amendment: December 14, 2000.

Brief description of amendment: These amendments revise Technical Specifications Sections 4.7.7.1.d.1 and 4.7.7.2.a. These changes increase the specified minimum number of compressed bottles of air from 84 to 102, and revise the differential pressure limit across the Control Room Emergency Ventilation System HEPA Filter, demister filter, and charcoal adsorber.

Date of issuance: December 12, 2001.

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

Amendment Nos.: 228 and 209.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 24, 2001 (66 FR 7687).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 12, 2001.

No significant hazards consideration comments received: No.

(Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

Dated at Rockville, Maryland, this 17th day of December, 2001.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 01–31473 Filed 12–21–01; 8:45 am]

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OFFICE OF MANAGEMENT AND BUDGET

Cost of Hospital and Medical Care Treatment Furnished by the United States; Certain Rates Regarding Recovery From Tortiously Liable Third Persons

By virtue of the authority vested in the President by section 2(a) of Public Law 87–693 (76 Stat. 593; 42 U.S.C. 2652), and delegated to the Director of the Office of Management and Budget by Executive Order No. 11541 of July 1, 1970 (35 FR 10737), the two sets of rates outlined below are hereby established. These rates are for use in connection with the recovery, from tortiously liable third persons, of the cost of hospital and medical care and treatment furnished by the United States (Part 43, Chapter I, Title 28, Code of Federal Regulations) through three separate Federal agencies. The rates have been established in accordance with the requirements of OMB Circular A–25, requiring reimbursement of the full cost of all services provided and will remain in effect until further notice. The rates for VA that were published in the **Federal Register** on October 31, 2000 remain in effect until further notice. The rates are as follows:

1. Department of Defense

The Department of Defense (DoD) reimbursement rates for inpatient, outpatient, and other services are provided in accordance with Title 10, United States Code, section 1095. Due to size, the sections containing the Drug Reimbursement Rates (section III.D.) and the rates for Ancillary Services Requested by Outside Providers (section III.E.) are not included in this package. Those rates are available from the TRICARE Management Activity's Uniform Business Office web site: *http://www.tricare.osd.mil/ebc/rm_home/imcp/ubo/ubo_01.htm.* The medical and dental service rates in this package (including the rates for ancillary services and other procedures requested by outside providers) are effective October 1, 2001. Pharmacy rates are updated on an as needed basis.

2. Health and Human Services

The tortiously liable rates for Indian Health Service health facilities are based on Medicare cost reports. The obligations for the Indian Health Service hospitals participating in the cost report project were identified and combined with applicable obligations for area offices costs and headquarters costs. The hospital obligations were summarized for each major cost center providing medical services and distributed between inpatient and outpatient. Total inpatient costs and outpatient costs were then divided by the relevant workload statistic (inpatient day, outpatient visit) to produce the inpatient and outpatient rates. In calculation of the rates, the Department's unfunded retirement liability cost and capital and equipment depreciation costs were incorporated to conform to requirements set forth in OMB Circular A–25.

In addition, the obligations for each cost center include obligations from certain other accounts, such as Medicare and Medicaid collections and the Contract Health fund, that were used to support the inpatient and outpatient workload. Obligations were excluded for certain cost centers that primarily support workloads outside of the directly operated hospitals or clinics (public health nursing, public health nutrition, health education). These obligations are not a part of the traditional cost of hospital operations and do not contribute directly to the inpatient and outpatient visit workload.

Separate rates per inpatient day and outpatient visit were computed for Alaska and the rest of the United States.