the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of the continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the petition for leave to intervene and request for hearing should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to General Counsel, Tennessee Valley Authority, ET 11A, 400 West Summit Hill Drive, Knoxville, TN 37902, attorney for the licensee.

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

For further details with respect to this action, see the application for amendment dated September 8, 2003, and supplement dated September 11, 2003, which are available for public inspection at the Commission's PDR, located at One White Flint North, File Public Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 12th day of September 2003.

For the Nuclear Regulatory Commission.

Margaret H. Chernoff,

Project Manager, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 03–23841 Filed 9–17–03; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, August 22, 2003, through September 4, 2003. The last biweekly notice was published on September 2, 2003 (68 FR 52233).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By October 16, 2003, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: August 19, 2003.

Description of amendment request: The amendments would revise the Technical Specifications (TS) to modify the requirements for the containment pressure control system to eliminate a problem with circuit fluctuation as a result of electronic noise.

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

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

The proposed amendment has no impact on any accident probabilities or consequences. The CPCS [containment pressure control system] functions to control the operation of the Containment Spray System and the Air Return System following certain design basis accidents. It cannot initiate any accidents by itself. Therefore, accident probabilities will be unaffected. Since the proposed change has been shown to have no effect upon any safety analysis results, the consequences of accidents will also be unaffected.

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

As stated previously, the CPCS in and of itself cannot initiate any accident condition. No change to any method of plant operation is being proposed in conjunction with this amendment request. Therefore, no new accident types can be created.

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

The proposed amendment will have no impact on any safety margin. None of the results of any existing safety analyses is affected as a result of the proposed change. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions. The fission product barriers include the fuel cladding, the reactor coolant pressure boundary, and the containment. None of these fission product barriers will be affected as a result of the proposed change. Therefore, no safety margin will be impacted.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, and Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, located in Mecklenburg County, North Carolina and York County, South Carolina

Date of amendment request: March 24, 2003, as supplemented June 25, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) to relocate reactor coolant system cycle-specific parameter limits from the TS to the core operating limits reports for the Catawba and the McGuire Nuclear Stations.

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:

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment does not involve a significant hazard.

Conformance of the proposed amendment to the standards for a determination of no significant hazards, as defined in 10 CFR 50.92, is shown in the following:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The relocation of Reactor Coolant System (RCS) related cycle-specific parameter limits from the Technical Specifications (TS) to the Core Operating Limits Reports (COLR) proposed by this amendment request does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSARS. Therefore, the proposed amendment does not result in the increase in the probability or consequences of an accident previously evaluated.

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

No. This change does not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of NRC approval of this amendment request. No changes are being made to the facility which should introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators.

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

No. Implementation of this amendment would not involve a significant reduction in the margin of safety. Previously approved methodologies will continue to be used in the determination of cycle-specific core operating limits appearing in the COLRS. Additionally, previously approved RCS minimum total flow rates for McGuire and Catawba are retained in their respective TS so as to assure that lower flow rates will not be used without prior NRC approval. Consequently, no safety margins will be impacted.

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

Attorney for licensee: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: August 11, 2003.

Description of amendment request: The proposed amendment would relocate Technical Specification (TS) Surveillance Requirement 4.5.2.f (vacuum leak rate test of the watertight enclosure for decay heat removal system valves DH–11 and DH–12) from the TSs to the Technical Requirements Manual.

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

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

Response: No.

Under the proposed change, initial conditions and assumptions remain as previously analyzed for accidents in the Davis-Besse Nuclear Power Station Updated Safety Analysis Report. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

Under the proposed change, the manner in which the watertight enclosure is sealed and tested is not altered, and the operability requirements of the watertight enclosure for Decay Heat Removal System valves DH–11 and DH–12 will continue to be adequately addressed by testing. No different accident initiators or failure mechanisms are introduced by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

Since there are no new or significant changes to the initial conditions contributing to accident severity or consequences, there are no significant reductions in a margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

GPU Nuclear Inc., Docket No. 50–320, Three Mile Island Nuclear Generating Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: July 21, 2003.

Description of amendment request: The amendment application proposes a revision to the Technical Specifications (TS) administrative controls for the radioactive effluent controls program. The proposed changes will make the Three Mile Island Nuclear Generating Station Unit 2 (TMI–2) radioactive effluent controls program technical specifications consistent with the technical specifications for the operating facility on site—Three Mile Island Nuclear Generating Station, Unit 1 (TMI–1). The proposed change adopts the TMI–1 liquid discharge limits since both TMI–1 and TMI–2 use the same liquid discharge monitor and have a common discharge pathway. The gaseous discharge limits will also be updated to reflect the current 10 CFR part 20 nomenclature along with some minor editorial changes. Additionally, the definition of a member of the public will be made consistent with the definition in 10 CFR part 20.

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

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

No. The TMI–2 TS for radioactive liquid effluent release, TS 6.7.4.a.2, will be revised to be consistent with the equivalent TS for TMI–1 (TS 6.8.4.b.(2)). The change will allow up to 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. Making the limits on the liquid effluent release concentrations for TMI–2 equivalent to those for TMI–1 is justified in that both units share a common effluent monitoring instrument and a common discharge path to the Susquehanna River.

The TMI–2 TS for limits on dose rate for radioactive gaseous effluent, TS 6.7.4.a.7, will be changed from the limits in 10 CFR 20, Appendix B, Table 2, Column 1, to be consistent with the equivalent TS for TMI–1 (TS 6.8.4.b.(7)). The revised limits will be as follows: (a) For noble gases: less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and (b) For tritium and all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ. The TMI–2 TS will continue to specify that annual and quarterly doses conform to Appendix I of 10 CFR Part 50.

The other changes are administrative and do no affect plant systems.

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

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

No. These changes will affect administrative controls on radionuclides that may be released from the site. It does not change the allowable off-site dose limits for any calendar year of operations. It does not change any plant system or the ALARA philosophy on discharges. Therefore, the proposed changes do not involve the possibility of a new or different type of accident from any accident previously evaluated.

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

No. These changes will affect the administrative controls on radionuclides that may be released from the site. It does not change the allowable off-site dose limits for any calendar year of operations. It does not change any plant system and will not affect the actual discharges from the plant. Therefore, there cannot be 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: Mary E. O'Reilly, Esq., First Energy Legal Department, 76 South Mail Street, Akron, OH 44308.

NRC Section Chief: Scott W. Moore.

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

Date of amendment request: August 28, 2003 (superseded the July 18, 2003, application).

Description of amendment request: The proposed amendment will increase the licensed power level to 1524 megawatts thermal (MWt) or 1.60 percent greater than the current power level of 1500 MWt. The requested increase in licensed rated power is the result of a measurement uncertainty recapture (MUR) power uprate. The information provided in support of this request is based on the NRC's Regulatory Issue Summary 2002–03, ''Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,'' dated January 31, 2002.

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.

There are no changes as a result of the MUR power uprate to the design or operation of the plant that could affect system, component, or accident functions. All systems and components function as designed and the performance requirements have been evaluated and found to be acceptable. The reduction in power measurement uncertainty allows for safety analyses to continue to be used without modification. This is because the safety analyses dependent on power level were performed or evaluated at 102% of 1500 MWt (1530 MWt) or higher. Analyses at these power levels support a core power level of 1524 MWt with a measurement uncertainty of 0.4%. Radiological consequences of USAR [Updated Safety Analysis Report] Chapter 14 accidents were assessed previously using the alternate source term methodology (Reference 10.2 [of the August 28, 2003, application]). These analyses were performed at 102% of 1500 MWt (1530 MWt) and continue to be bounding. Updated Safety Analysis Report (USAR) Chapter 14 analyses and accident analyses continue to demonstrate compliance with the relevant accident analyses' acceptance criteria.

Therefore, there is no significant increase in the consequences of any accident previously evaluated.

The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) were evaluated at an uprated core power level of 1524 MWt and continue to comply with their applicable structural limits. These analyses also demonstrate the components will continue to perform their intended design functions. Changing the heatup and cooldown curves is based on uprated fluence values. This does not have a significant effect on the reactor vessel integrity. Thus, there is no significant increase in the probability of a structural failure of the primary loop components. The LBB [leak-before-break] analysis conclusions remain valid and the breaks previously exempted from structural consideration remain unchanged.

All of the NSSS [nuclear steam system supplier] systems will continue to perform their intended design functions during normal and accident conditions. The auxiliary systems and components continue to comply with the applicable structural limits and will continue to perform their intended functions. The NSSS/BOP [balance-of-plant] interface systems were evaluated at 1524 MWt and will continue to perform their intended design functions. Plant electrical equipment was also evaluated and will continue to perform their intended functions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function at the uprated power level. The proposed change has no adverse effects on any safety related systems or component and does not challenge the performance or integrity of any safety related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Operation at 1524 MWt core power does not involve a significant reduction in the margin of safety. The current accident analyses have been previously performed with a 2% power measurement uncertainty or at uprated core powers that exceed the MUR uprated core power. System and component analyses have been completed at the MUR uprated core power conditions. Analyses of the primary fission product barriers at uprated core powers have concluded that all relevant design basis criteria remain satisfied in regard to integrity and compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been both reviewed and approved by the NRC, or are currently under review (the proposed Pressure-Temperature Limits Report). Therefore, the proposed change does not involve a significant reduction in margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

Southern California Edison Company, et al., Docket No. 50–206, San Onofre Nuclear Generating Station, Unit 1, San Diego County, California

Date of amendment requests: July 25, 2003.

Description of amendment requests: The amendment application requests a revision to the Unit 1 Defueled Safety Analysis Report (DSAR) that concerns the turbine gantry crane, turbine gantry crane capacity, fuel shipment and the structural descriptions of the turbine building. The licensee is engineering structural changes to the turbine building and gantry crane and replacing the turbine gantry crane trolley in preparation for moving spent fuel from the Unit 1 spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI). With the planned modifications listed above, the licensee will be able to satisfy the guidance of NUREG–0612, ''Control of Heavy Loads at Nuclear Power Plants,'' and NUREG–0554. ''Single-Failure Proof Cranes for Nuclear Power Plants,'' (regarding safe load handling paths and single-failure proof cranes) in performing the necessary movement of Unit 1 spent fuel to dry cask storage.

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

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

No. The DSAR addresses fuel handling accidents. The process for transporting a cask is essentially unchanged from that previously performed. The building arrangement is such that the cask is never carried over the spent fuel pool. The transport height of the cask has been increased to a minimum of 9 inches based on the design of the new Ederer X-Sam single-failure proof trolley. Because the turbine gantry crane upgrade improves the reliability of the crane, a single failure will not result in loss of its capability to safely retain control of the hook load.

If a portion of the new turbine gantry crane lifting device malfunctions or fails, the crane system is designed such that the load will move a limited distance downward prior to backup restraints becoming engaged. The increased minimum transport height (9 inches) is established to accommodate the design features. The probability of a fuel handling accident is unchanged. Because the spent fuel fission product activity has decayed by more than ten years compared to the source term analyzed in the DSAR, the consequences of the analyzed fuel handling accident are significantly lessened.

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

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

No. By implementing use of a qualified single-failure proof crane for cask handling, accidental dropping of the cask is not postulated. The cask load will be increased to a maximum of 105 tons under the new single failure proof turbine gantry crane design. The construction of a single failure proof turbine gantry crane mitigates the potential for an accident, since a single failure will not result in the loss of its capability to safely retain control of the hook load.

Therefore, performing fuel transfer in a manner consistent with the proposed DSAR amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed DSAR change makes use of analysis methods and inputs consistent with other structural and safety analyses given in the DSAR. The turbine gantry crane will be upgraded to comply with the single failure proof requirements of NUREG–0554. The safety margins provided by the new crane design have either remained the same or have been enhanced to ensure adequate margin to prevent failure of the crane or any lifting devices associated with the lifting of a spent fuel transfer cask.

Therefore, the proposed DSAR 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: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Scott W. Moore.

Southern California Edison Company, et al., Docket Nos. 50–361 and 50–362, San Onofre Nuclear Generating Station, Units 2 and 3, San Diego County, California

Date of amendment requests: August 4, 2003.

Description of amendment requests: The proposed amendments would revise Technical Specification 3.9.3, ''Containment Penetrations.'' Specifically, a Note will be added to the Limiting Condition for Operations that permits the Containment equipment hatch to be open during core alterations and movement of irradiated fuel in containment.

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

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

Response: No.

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

The proposed change to Technical Specification 3.9.3 would allow the containment equipment hatch to be open during fuel movement or core alterations. Currently, the equipment hatch is closed with four bolts during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. The containment equipment hatch is not an initiator of an accident. Whether the containment equipment hatch is open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated.

Allowing the containment equipment hatch to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR Part 100 and the calculated control room operator dose are within the limits of 10 CFR [Part] 50 Appendix A General Design Criterion (GDC) 19. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate containment closure after the containment is evacuated, which is much less than the two hours assumed in the analysis. The proposed change should significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment.

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

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

Response: No.

The proposed change to Technical Specification 3.9.3, ''Containment Penetrations,'' affects a previously evaluated fuel handling accident inside containment. The new Fuel Handling Accident analysis continues to assume that all of the iodine and noble gases that become airborne escape the containment within two hours, and reach the exclusion area boundary and control room with no credit taken for containment air exhaust filtration, or for decay or deposition during atmospheric dispersion. The change will include the addition of flashing that will restrict a release of post-accident fission products when the Containment Structure Equipment Hatch Shield Doors are in their closed position. In this manner, the closed Shield Doors will provide Containment closure. Accordingly, since the proposed change does not functionally alter the design of plant systems and the revised analysis is consistent with the Fuel Handling Accident analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. [The containment equipment hatch is not an initiator of an accident.]

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

Response: No.

The margin of safety as defined by 10 CFR Part 100 has not been significantly reduced. The calculated dose is well within the limits given in 10 CFR Part 100 as defined by Standard Review Plan 15.7.4. The analysis does not credit closing the Containment Structure Equipment Hatch Shield Doors. Accordingly, the proposed change does not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussion, Southern California Edison has determined that the proposed amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety, therefore, the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92.

Therefore, the operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California

Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260 and 50–296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: July 25, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.1.8, ''Scram Discharge Volume (SDV) Vent and Drain Valves,'' to allow a vent or drain line with one inoperable valve to be isolated instead of requiring the valve to be restored to Operable status within 7 days.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18295). The licensee affirmed the applicability of the model NSHC determination in its application dated July 25, 2003.

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.

A change is proposed to allow the affected SDV vent and drain line to be isolated when there are one or more SDV vent or drain lines with one valve inoperable instead or requiring the valve to be restored to operable status within 7 days. With one SDV vent or drain valve inoperable in one or more lines, the isolation function would be maintained since the redundant valve in the affected line would perform its safety function of isolating the SDV. Following the completion of the required action, the isolation function is fulfilled since the associated line is isolated. The ability to vent and drain the SDVs is maintained and controlled through administrative controls. This requirement assures the reactor protection system is not adversely affected by the inoperable valves. With the safety functions of the valves being maintained, the probability or consequences of an accident previously evaluated are not significantly increased.

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

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

The proposed change ensures that the safety functions of the SDV vent and drain valves are fulfilled. The isolation function is maintained by redundant valves and by the required action to isolate the affected line. The ability to vent and drain the SDVs is maintained through administrative controls. In addition, the reactor protection system will prevent filling of an SDV to the point that it has insufficient volume to accept a full scram. Maintaining the safety functions related to isolation of the SDV and insertion of control rods ensures that the proposed change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

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

Date of amendment request: August 22, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System Instrumentation.'' The revision adds a Surveillance Requirement for response time to the Source Range (SR) Neutron Flux Reactor Trip function.

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

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

No. The proposed amendment enhances the operability of the SR reactor trip channels by requiring response time testing. This will provide additional assurance that the plant will be operated within its design and licensing basis. The change does not involve any physical modifications or functional design changes to the SR instrumentation, and will not alter any system interfaces. The design standards, criteria, and material specifications applicable to the design and installation of the SR instrumentation still apply. The performance of response time testing for the SR Neutron Flux channels does not contribute to the initiation of any accident previously evaluated. Testing will be performed when the SR reactor trip function is not required to be operable. A response time will ensure that a Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from Subcritical (RWFS) event in Modes 3, 4, or 5 remains bounded by the current analysis and the reactor would be shutdown before any significant power is generated. Thus, the probability of occurrence of an accident evaluated in the Updated Final Safety Analysis Report (UFSAR) will not increase as a result of the performance of response time testing. The performance of response time testing will not affect any radiological barriers. The testing will not alter any operator responses required for accident mitigation and will not change any assumptions made in evaluating radiological consequences of an accident described in the UFSAR. The consequences of an RWFS event occurring from Mode 3, 4, or 5 are less severe than from Mode 2 since reactivity levels are lower in the lower modes. Therefore, there is no potential for an increase in the consequences of any previously evaluated accident.

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

No. The proposed change will not require any changes to hardware, setpoints, or design functions. The addition of a response time test requirement will not change the way the system is operated but will impose more restrictive operability requirements for the SR reactor trip function. This enhancement to the operability requirements for a protection system function is not considered an accident initiator. Therefore, the activity will not create a new or different kind of accident from those previously evaluated in the UFSAR.

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

No. The proposed change does not involve any changes to setpoints or safety limits. The required response time is consistent with the current accident analysis described in UFSAR and will ensure that a RWFS event in Modes 3, 4, or 5 remains bounded by the current analysis. The addition of a response time verification requirement is an enhancement to the operability requirements of the SR reactor trip channels and does not reduce 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Section Chief: Allen G. Howe.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action *see* (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application of amendments: April 10, 2003, as supplemented by letter dated July 1, 2003.

Brief description of amendments: The amendments revised frequencies associated with the Technical Specification Surveillance Requirements 3.4.12.5 and 3.4.12.7 concerning the Low Temperature Overpressure Protection System.

Date of Issuance: August 25, 2003.

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

Amendment Nos.: 333, 333, and 334.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 27, 2003 (68 FR 2885).

The supplement dated July 1, 2003, provided clarifying information that did not change the scope of the April 10, 2003, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 25, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 30, 2003, as supplemented by letters dated August 1 and 12, 2003.

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

Date of issuance: September 3, 2003.

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

Amendment No.: 250.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 22, 2003 (68 FR 43384).

The August 1 and 12, 2003, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

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

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: August 23, 2002, as supplemented July 2, 2003.

Brief description of amendments: The amendments revise (1) the Operating Licenses to delete obsolete and expired license conditions and make administrative and editorial changes, and (2) the Technical Specifications (TSs) to make administrative and editorial changes.

Additionally, the licensee proposed to delete the radiation monitoring instrumentation identification numbers from certain TSs. The licensee will be submitting new information to support these changes in a future request. The NRC staff will handle this request under separate cover.

Date of issuance: August 22, 2003.

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

Amendment Nos.: 279 and 261.

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

*Date of initial notice in **Federal Register:*** October 15, 2002 (67 FR 63695).

The supplement dated July 2, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 2003.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: May 2, 2003, as supplemented by letters dated June 30, July 30, August 8, and 18, 2003.

Brief description of amendment: The amendment updates the existing reactor coolant system pressure and temperature limit curves (TS Figure 3.4.9–1) and extends their applicability to 32 effective full power years.

Date of issuance: August 25, 2003.

Effective date: As of the date of issuance and shall be implemented by September 1, 2003.

Amendment No.: 253.

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

Date of initial notice in ***Federal Register:*** May 27, 2003 (68 FR 28855).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 25, 2003.

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: January 29, 2003.

Brief description of amendment: The amendment revises the drywell leakage and sump monitoring detection section of the current Technical Specifications (TSs). Specifically, the changes clarify the associated definitions and divide TS 3.6.D/4.6.D, ''Coolant Leakage,'' into two subsections and retitle it ''Reactor Coolant System (RCS).'' One of the subsections contains the Limiting Condition for Operations (LCOs) for RCS operational leakage, and the other subsection contains the LCOs for the RCS leakage detection instrumentation.

Date of issuance: August 21, 2003.

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

Amendment No.: 137.

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

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18279).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 21, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 27, 2002, and its supplements dated May 15, June 26, and August 1, 2003.

Brief description of amendments: The amendments revised Table 3.3.1–1, ''Reactor Trip System Instrumentation'' of the technical specifications to replace the term ''minimum measured flow per loop'' to ''measured loop flow'' in the allowable value and nominal trip setpoint for the reactor coolant flow-low reactor trip function, and delete footnote (l). The amendments also allow an alternate method for the measurement of reactor coolant system (RCS) total volumetric flow rate through measurement of the elbow tap differential pressure on the RCS primary cold legs.

Date of issuance: August 21, 2003.

Effective date: August 21, 2003, and shall be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1–161; Unit 2–162.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 7, 2003 (68 FR 810).

The May 15, June 26, and August 1, 2003, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 21, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 3, 2001, as supplemented August 7, 2001, October 29, 2001, May 3, 2002, October 7, 2002, November 5, 2002, and June 6, 2003.

Brief description of amendment: The amendment revised the Ginna Station Improved Technical Specifications to reflect design changes to the actuation circuitry associated with the Control Room Emergency Air Treatment System (CREATS). The proposed design changes consist of replacing the current diverse radiation monitors with two Geiger-Mueller (GM) tubes powered from two separate safety-related power supplies which are configured into two redundant actuation logic trains using safety-grade digital instrumentation. The design changes are intended to increase system reliability by providing redundancy and reducing spurious actuations. The amendment changes limiting condition for operation 3.3.6 for the CREATS Actuation Instrumentation as follows:

a. Adds a new Condition to require immediately placing the CREATS in the emergency mode of operation upon the loss of two instrument channels/trains.

b. Adds a new surveillance requirement involving a CHANNEL CHECK of the Control Room Radiation Intake Monitors.

c. Revises Table 3.3.6–1 to increase the number of trains of Manual and Automatic Initiation Circuits from one train to two trains.

d. Extends the Completion Time of the Required Action for a loss of one channel/train from 1 hour to 7 days as the result of installing redundant channels/trains.

e. Revises Table 3.3.6–1 to remove reference to the iodine, noble gas, and particulate control room radiation intake monitors. These monitors will be replaced by the two new GM tubes.

f. Revises Table 3.3.6–1 to replace the column heading ''Trip Setpoint'' with ''Allowable Value.''

Date of issuance: August 29, 2003.

Effective date: August 29, 2003.

Amendment No.: 83.

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

Date of initial notice in ***Federal Register***: September 5, 2001 (66 FR 46481).

The supplemental letters referenced above provided clarifying information that did not change the scope of the amendment as described in the original notice, and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2003.

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: September 24, 2002, as supplemented by letters dated May 20 and July 16, 2003.

Brief Description of amendments: The changes revise Technical Specifications

(TS) 3.7.10, ''Control Room Emergency Filtration/Pressurization System (CREFS),'' and TS 3.7.12 ''Penetration Room Filtration (PRF) System,'' to establish actions to be taken for inoperable ventilation systems due to a degraded control room pressure boundary or PRF and spent fuel pool room boundary, respectively. This revision approves changes that would allow up to 24 hours to restore the pressure boundary to an operable status when two ventilation trains are inoperable due to an inoperable pressure boundary in MODES 1, 2, 3, and 4. In addition, a Limiting Condition for Operation Note would be added to allow the pressure boundary to be opened intermittently under administrative control without affecting CREFS or PRF System operability. The applicable TS Bases have been revised to document the TS changes and to provide supporting information. These changes are based on Technical Specifications Task Force document TSTF–287, Revision 5.

Date of issuance: August 22, 2003.

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

Amendment Nos.: 161 and 154.

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

*Date of initial notice in **Federal Register:*** November 12, 2002 (67 FR 68744).

The supplements dated May 20 and July 16, 2003, provided clarifying information that did not change the scope of the September 24, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 22, 2003.

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: May 14, 2003.

Brief description of amendments: The amendments revise Surveillance.

Requirement 4.6.2.1 for demonstrating operability of containment spray system spray nozzles to require verification of operability only after spray ring header maintenance that could result in nozzle obstructions without specifying the method of verification.

Date of issuance: August 20, 2003.

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

Amendment Nos.: Unit 1–156; Unit 2–144.

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

*Date of initial notice in **Federal Register***: June 24, 2003 (68 FR 37582).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 20, 2003.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–259, 50–260, and 50–296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: April 15, 2003.

Description of amendment request: The amendments revised Technical Specification (TS) 3.7.3, ''Control Room Emergency Ventilation (CREV) System,'' to allow up to 24 hours to restore the control room pressure boundary (CRPB) to operable status when two trains of the ventilation system are inoperable due to an inoperable CRPB in MODES 1, 2, and 3. In addition, a note is included to allow the pressure boundary to be opened intermittently under administrative controls without affecting the CREV System operability. The licensee revised the applicable TS Bases to make them consistent with the TS changes. These changes are based on TS Task Force Traveler No. 287, which was approved by the NRC on March 16, 2000.

Date of issuance: August 29, 2003.

Effective date: Date of issuance, to be implemented within 60 days.

Amendment Nos.: 246, 283 and 241.

Facility Operating License Nos. DPR–33, DPR–52, and DPR–68. Amendments revised the TSs.

*Date of initial notice in **Federal Register:*** May 27, 2003 (68 FR 28858).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2003.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 5, 2003.

Brief description of amendments: The amendments extend from 1 hour to 24 hours the completion time for Condition B of Technical Specification 3.5.1, which defines requirements for the restoration of an emergency core cooling system accumulator when it has been declared inoperable for a reason other than boron concentration.

Date of issuance: August 25, 2003.

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

Amendment Nos.: 106 and 106.

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

*Date of initial notice in **Federal Register:*** July 8, 2003 (68 FR 40721).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 25, 2003.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: June 9, 2003, as supplemented on July 28, 2003.

Brief Description of amendments: These amendments revise Section 6 of the Surry Power Station Technical Specifications (TS) for Units 1 and 2 to adopt the format for topical report references that are described in Industry/Technical Specifications Task Force Traveler, TSTF–363, Rev 0, ''Revised Topical Report References in Improved Technical Specification (ITS) 5.6.5, [Core Operating Limits Report] COLR.''

Date of issuance: August 27, 2003.

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

Amendment Nos.: 235 and 234.

Renewed Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 8, 2003 (68 FR 40722).

The July 28, 2003, supplement contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 27, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 5th day of September 2003.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 03–23251 Filed 9–17–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–346; License No. NPF–03]

FirstEnergy Nuclear Operating Company; Notice of Issuance of Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, has issued a Director's Decision with regard to a letter dated February 3, 2003, filed by Congressman Dennis Kucinich, Representative for the 10th Congressional District of the State of Ohio in the United States House of Representatives, hereinafter referred to as the ''petitioner.'' The petition was supplemented on March 27, 2003. The petition concerns the operation of the Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse), located in Ottawa County, Ohio.

The Petitioner requested that the U.S. Nuclear Regulatory Commission (NRC) immediately revoke the FirstEnergy Nuclear Operating Company's (FENOC's or the licensee's) license to operate the Davis-Besse Nuclear Power Station, Unit 1 (Davis-Besse), located in Ottawa County, Ohio. As an alternative, the Petitioner asked the NRC to reexamine its denial of a previous 2.206 petition, submitted by the Toledo Coalition for Safe Energy *et al.*, that requested the NRC issue an order to the licensee requiring a verification by an independent party for issues related to the reactor vessel head damage at Davis-Besse.

The basis for the request was that FENOC ''has operated outside the parameters of their operating license for several years, has violated numerous federal laws, rules and regulations, and has hidden information from the NRC and lied to the NRC to justify the continuing operation of the Davis-Besse Nuclear Power Station.'' The Petitioner supported his request by citing various publicly available documents and information related to reactor pressure vessel head damage discovered at Davis-Besse in March 2002. The documents describe noncompliance with the Davis-Besse operating license and violations of NRC regulations. The documents include NRC inspection reports, newspaper articles, and reports published by the Union of Concerned scientists.

By an acknowledgment letter dated February 10, 2003, the NRC staff formally notified the Petitioner that the letter dated February 3, 2003, met the criteria for review under 10 CFR 2.206, and that the NRC staff would act on the request within a reasonable time. The acknowledgment letter further stated that the Davis-Besse facility was shut down, and would remain so, until the NRC is satisfied that there is reasonable assurance of adequate protection of the public health and safety and that issues associated with management of the facility and potential wrongdoing have been satisfactorily addressed. The NRC staff also informed the Petitioner in the acknowledgment letter that the issues raised in the petition were being referred to NRR for appropriate action.

On March 27, 2003, the Petitioner submitted supplemental information to support the petition. The licensee responded to the Petition on February 27, 2003, and to the supplement on April 11, 2003. These responses were considered by the staff in its evaluation of the petition. Copies of the licensee's responses are publicly available in the NRC's NRC's Agencywide Documents Access and Management System (ADAMS).

The NRC sent a copy of the proposed Director's Decision to the Petitioner and to licensee for comment on June 6, 2003. The Petitioner and FENOC both responsed with comments on July 7, 2003. The comments and the NRC staff's response to them are included with the Director's Decision.

The Director of the Office of Nuclear Reactor Regulation has determined that the request to revoke the Davis-Besse operating license and the alternative request for the NRC to reexamine its denial of a previous 2.206 petition, submitted by the Toledo Coalition for Safe Energy *et al.*, that requested the NRC issue an order to the licensee requiring a verification by an independent party for issues related to the reactor vessel head damage at Davis-Besse, both be denied. The reasons for these decisions are explained in the Director's Decision pursuant to 10 CFR 2.206 DD–03–03, the complete text of which is available in ADAMS, or are available for inspection at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records are accessible from the ADAMS Public Electronic Reading Room on the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC PDR reference staff at 1–800–397–4209 or 301–415–4737, or by e-mail to *pdr@nrc.gov.*

The NRC staff has carefully considered the Petitioner's arguments regarding why FENOC's operating license for the Davis-Besse Nuclear Power Station should be revoked, as well as the alternative request for verification by independent party. The NRC staff shares the Petitioner's concerns about verifying the adequacy of plant operator performance and ensuring that future operation of the plant is conducted safely and in compliance with NRC requirements. The licensee has established, and is implementing, a Return-to-Service Plan that comprehensively addresses human factors, programmatic, and equipment issues along with the specific corrosion of the reactor vessel head. This includes evaluating, testing, or inspecting plant safety-related systems to ensure that they are able to perform their design-basis functions as defined in the plant's technical specifications and Updated Final Safety Analysis Report. Additionally, the NRC's has implemented enhanced oversight of the Davis-Besse facility that included the creation of an oversight panel to provide the required oversight during the plant shutdown, any future restart, and following restart until a determination is made that the plant is ready for return to the NRC's normal Reactor Oversight Process. The NRC's inspection activities go beyond ensuring that the direct causes of the damage to the reactor vessel head are properly identified and corrected. The NRC's activities also look broadly at safety-related plant systems and programs to ensure that the physical condition of the plant is adequate and the licensee's operations, maintenance, and engineering organizations are prepared to operate the plant safely if it is permitted to restart. Thus the NRC believes that the FENOC Return-to-Service Plan, as monitored by the NRC Davis-Besse Oversight Panel, provides an appropriate opportunity for FENOC to demonstrate or achieve compliance with NRC requirements, and that these activities will provide results that adequately address the Petitioner's stated safety concerns.

With regard to the specific punitive action of revoking the Davis-Besse operating license sought by the Petitioner, the NRC staff finds that there is insufficient basis to take the requested action. While serious violations did occur at the Davis-Besse facility, the violations in and of themselves do not