Week of August 5, 2002—Tentative

There are no meetings scheduled for the Week of August 5, 2002.

Week of August 12, 2002—Tentative

Tuesday, August 13, 2002

9:30 a.m. Briefing on Special Review Group Response to the Differing Professional Opinion/Differing Professional View (DPO/DPV) Review (Public Meeting) (Contact: John Craig, 301–415–1703).

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

Week of August 19, 2002—Tentative

Wednesday, August 21, 2002

9:30 a.m. Briefing on NRC International Activities (Public Meeting) (Contact: Janice Dunn Lee, 301–415–1780).

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

2 p.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: John Zabko, 301–415–2308).

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

Week of August 26, 2002—Tentative

There are no meetings scheduled for the Week of August 26, 2002.

* The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292.

CONTACT PERSON FOR MORE INFORMATION: David Louis Gamberoni (301) 415–1651.

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ADDITIONAL INFORMATION: By a vote of 4–0 on July 12, the Commission determined pursuant to U.S.C. 552b(e) and § 9.107(a) of the Commission's rules that ''Discussion of Intragovernmental Issues (Closed—Ex. 9)'' be held on July 12, and on less than one week's notice to the public.

* * * * *

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

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

Dated: July 18, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–18727 Filed 7–14–02; 1:16 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 28, 2002, through July 11, 2002. The last biweekly notice was published on July 9, 2002 (67 FR 45560).

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, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By August 22, 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,[1]

[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002,

Continued

which is available at the Commission's PDR, 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 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, 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, 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, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

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inadvertently omitted the last sentence of 10 CFR 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: "In all other circumstances, such ruling body or officer shall, in ruling on—

(1) A petition for leave to intervene or a request for hearing, consider the following factors, among other things:

(i) The nature of the petitioner's right under the Act to be made a party to the proceeding.

(ii) The nature and extent of the petitioner's property, financial, or other interest in the proceeding.

(iii) The possible effect of any order that may be entered in the proceeding on the petitioner's interest.

(2) The admissibility of a contention, refuse to admit a contention if:

(i) The contention and supporting material fail to satisfy the requirements of paragraph (b)(2) of this section; or

(ii) The contention, if proven, would be of no consequence in the proceeding because it would not entitle petitioner to relief."

Dominion Nuclear Connecticut, Inc., Docket No. 50–245, Millstone Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: May 13, 2002.

Description of amendment request: The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 1 (MP1) Permanently Defueled Technical Specifications (TSs) to change selected MP1 radiological related TSs. These changes are due to the revision to part 20 of Title 10 of the Code of Federal Regulations.

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

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

It is proposed to revise the Occupational Radiation Exposure Report, Radioactive Effluent Controls Program, and High Radiation Area Specifications in accordance with TSTF [Technical Specification Task Force] travelers 152, 258 and 308, to reflect changes due to the revision to 10 CFR part 20.

These changes do not have an impact on the acceptance criteria for any design basis accident described in the Unit No. 1 Defueled Safety Analysis Report (DSAR).

The changes have no impact on plant equipment operation. Since the changes are administrative or editorial in nature they cannot affect the likelihood or consequences of accidents. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

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

The revisions to the Occupational Radiation Exposure Report, Radioactive Effluent Controls Program, and High Radiation Area Specifications in accordance with TSTF travelers 152, 258 and 308 will have no effect on plant operation. Since the proposed changes are solely administrative or editorial in nature, they do not affect plant operation in any way.

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

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

Since the proposed changes are solely administrative or editorial changes to the TS, they do not affect plant operation in any way. The proposed changes to each unit's technical specifications will revise them to reflect the requirements of the current 10 CFR Part 20, standardize terminology, provide clearer guidance, clarify inconsistencies, remove extraneous information, and result in minor format changes that will not result in any technical changes to current requirements.

The proposed changes have no effect on any safety analyses assumptions and therefore [do] not impact any margins of safety. The proposed changes do not impact any acceptance criteria for the design basis accidents described in the Unit No. 1 DSAR and [do] not impact the consequences of accidents previously evaluated. Therefore, the proposed changes will not result in a reduction in a margin of safety.

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

Attorney for licensee: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, CT 06385.

NRC Section Chief: Stephen Dembek.

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: May 9, 2002.

Description of amendment request: The amendments would revise the licensing basis Steam Generator Tube Rupture sequences for Catawba Nuclear Station, Units 1 and 2. Specifically, it is requested that a certain single failure scenario potentially leading to steam generator overfill be excluded from the design basis steam generator tube rupture analysis using the guidance of Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis.''

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

No. This proposed amendment requests that steam generator tube rupture sequences involving a failure of 125 VDC Distribution Center EDE [or EDF] be excluded from consideration in the analysis of the design basis steam generator tube rupture event. These sequences involve a single failure that potentially degrades the ability to terminate auxiliary feedwater flow into a ruptured steam generator following a steam generator tube rupture. The inability to terminate auxiliary feedwater flow in a timely manner following a steam generator tube rupture could result in steam generator overfill.

The sequences to be excluded do not involve equipment that can be considered an accident initiator. Implementation of this amendment does not involve any physical changes to the facility. It does not affect basic operation of the facility. The probability of occurrence of a steam generator tube rupture or any other accident previously evaluated will not change following implementation of this amendment.

Elimination of certain sequences from the design basis steam generator tube rupture analysis does not adversely affect the ability to cool the reactor core and prevent core damage following a steam generator tube rupture. The Departure from Nucleate Boiling ratio is not adversely impacted.

The ability to maintain a secondary heat sink and provide water to the Reactor Coolant System for makeup, cooling of the core, and shutdown margin following a design basis steam generator tube rupture is not affected by the changes proposed in this license amendment. Neither fuel damage nor clad damage is expected to occur for the steam generator tube rupture sequences to be eliminated.

Should the ruptured steam generator overfill following a design basis steam generator tube rupture in one of the sequences to be excluded, radioactivity could be released to the environment in increased amounts and over a longer time span than predicted in the safety analysis. The frequency of occurrence of these steam generator tube rupture sequences is low. Should such an event occur, the radiological consequences are expected to be below the guidelines of 10 CFR 100 and General Design Criteria 19. Under nominal conditions, (e.g., nominal atmospheric dispersion factors, nominal levels of radioactivity in the Reactor Coolant System, etc.), radiological consequences of a steam generator tube rupture would be small compared to even the guideline values of the Standard Review Plan, Section 15.6.3. There is no significant adverse effect on the mitigation of consequences following a steam generator tube rupture.

In summary, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment involves elimination of certain sequences from the design basis steam generator tube rupture analysis. No physical changes to the facility are associated with the proposed amendment.

The sequences to be eliminated involve single failures that could adversely affect the ability to terminate auxiliary feedwater flow to a ruptured steam generator. The failures associated with these sequences are not accident sequence precursors and do not have an adverse impact on any accident initiator.

No new failure modes are created due to implementation of the change proposed in this License Amendment Request. Therefore, operation of the facility in accordance with the changes proposed in this License Amendment Request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. One of the standards by which the consequences of the design basis steam generator tube rupture are evaluated is that the Departure from Nucleate Boiling Ratio (DNBR) is greater than the limit value. Should one of the steam generator tube rupture sequences to be excluded occur, the effects relative to steam generator overfill would not be manifested until the Control Room operators attempt to stop the flow of auxiliary feedwater to the ruptured steam generator which is well into the event. The minimum DNBR would occur within seconds after reactor trip. Therefore, the criterion concerning DNBR is met.

The risk evaluation demonstrates that the frequency of steam generator overfill associated with the steam generator tube rupture sequences to be excluded is low (approximately 3.7 E–11 per reactor year per Class 1E Train). Additionally, the frequency of a large early release is shown to be very low (approximately 3.7 E–15 per reactor year per Class 1E Train).

It is concluded that removal of certain steam generator tube rupture sequences from the plant licensing basis as proposed does not constitute a significant reduction in a margin of safety.

Based on this evaluation, it is concluded that operation of the facility in accordance with the proposed amendment constitutes no significant hazard to the public.

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. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of amendment request: June 7, 2002.

Description of amendment request: The proposed amendments would revise the Updated Final Safety Analysis Report to eliminate credit for the flow path from the spent fuel pool to high pressure injection pump as one source of primary system makeup following a tornado. The proposed amendments would also credit the Standby Shutdown Facility as the assured means of achieving safe shutdown for all three Oconee units following a tornado.

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

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

The changes being requested in this amendment request involve (1) the elimination of the Spent Fuel Pool (SFP) as a suction source to a High Pressure Injection [HPI] pump for primary system make-up, and (2) to fully credit the Standby Shutdown Facility (SSF) as the primary assured means of achieving safe shutdown of all three units following a tornado. Following the modification to fully tornado protect the SSF, this facility becomes the station's assured flow path for both primary make-up and secondary decay heat removal for all three units.

Although the probability of a severe tornado strike at the station does not change, new tornado insights gained from a review of the current external event risk analysis have resulted in an enhanced risk model that more accurately characterizes station tornado damage risk. The proposed changes are part of the revised tornado mitigation strategy that provides for an assured, deterministic success path rather than the current strategy that is based on risk insights and diversity for achieving safe shutdown. This effort has resulted in an overall reduction in tornado risk at the station and consequently, would not result in a significant increase in the consequences of an accident previously evaluated.

Other than the fortification of walls of existing structures to harden them against tornado damage, there are no physical changes to the plant structures, systems, or components (SSCs) or operating procedures, nor are there any changes to safety limits or set points. Also, no new radiological release pathways are created.

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

The changes being proposed in this amendment request do not create the possibility of a new or different kind of accident from any accident previously evaluated. The initial placement of the SFP-HPI flow path into the LB (Licensing Bases) was based on 1989 risk analyses that showed a potential need for primary make-up due to inventory losses from a reactor coolant pump (RCP) seal loss-of-cooling accident (LOCA). The upgrade of the RCP seals has significantly reduced the probability of a seal LOCA and subsequently, alleviated the initial reliance on the SFP–HPI flow path for primary make-up. If multi-unit primary make-up and decay heat removal are required following an event, the tornado protected SSF RB[C]MU (Reactor Coolant Makeup) or SSF ASW (Auxiliary Service Water) pumps have the capabilities to perform these functions for all three units.

Other than the fortification of walls of existing structures to harden them against tornado damage, there are no physical changes to the plant SSCs or operating procedures. There are no new hazardous materials or potential missiles. It does not introduce the possibility of any new or different malfunctions. No safety limits or set points are changed.

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

As mentioned previously, new tornado insights gained from a review of the current external event risk analysis have resulted in an enhanced risk model that more accurately characterizes station tornado damage risk. The proposed changes are part of the revised tornado mitigation strategy that provides for an assured, deterministic success path rather than a strategy that is based on risk insights and diversity for achieving safe shutdown.

There are no safety limit, set point, design parameters, or operating procedure changes required. The integrity of the fuel cladding, reactor coolant system, and containment are preserved. Thus, 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: May 14, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours'' to ''* * * up to 24 hours or up to the limit of the specified interval, whichever is greater.'' In addition, the following requirement would be added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. 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 May 14, 2002.

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., 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: June 12, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period would be 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 SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. 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 June 12, 2002.

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 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.

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

Date of amendment request: May 31, 2001.

Description of amendment request: The proposed amendments would change Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change modifies TS Surveillance Requirement (SR) 3.6.1.3.8 to reduce to number of excess flow check valves (EFCVs) required to be tested every 24 months. The proposed SR will require that a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrumentation line break signal every 24 months. All reactor instrumentation line EFCVs will be tested at least once every 10 years (nominal). The proposed change implements Technical Specification Task Force Traveler 334 (TSTF–334), ''Relaxed Surveillance Frequency for Excess Flow Check Valve Testing,'' Revision 2.

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 change to LaSalle County Station, Unit 1 and Unit 2 Technical Specifications (TS) modifies TS Surveillance Requirement (SR) 3.6.1.3.8 to reduce the number of excess flow check valves (EFCVs) required to be tested every 24 months. The proposed SR will require that a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrumentation line break signal every 24 months. All reactor instrumentation line EFCVs will be tested at least once every 10 years (nominal).

The performance of EFCV surveillance testing is not a precursor to any accident previously evaluated and is not related to the frequency of instrument line failures. Thus, the proposed change to modify the test frequency associated with EFCV surveillance does not have any effect on the probability of an accident previously evaluated.

The performance of the EFCV surveillance testing does provide assurance that the EFCV will perform as designed. The LaSalle County Station radiological dose assessment for an instrument line break is documented in the LaSalle County Station UFSAR Table 15.6–4, ''Instrument Line Break Radiological Effects.'' The assessment does not credit performance of the EFCV to limit instrument line flows during an assumed break. These estimated doses are significantly below the regulatory dose limits listed in 10 CFR 100, ''Reactor Site Criteria.'' The proposed change does not change the assumptions or the estimated doses associated with a LaSalle County Station instrument line break. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the proposed change does 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 change modifies TS SR 3.6.1.3.8 to reduce the number of excess flow check valves (EFCVs) required to be tested every 24 months while requiring all EFCVs to be tested at least once every 10 years (nominal). The proposed change does not affect the performance of any LaSalle County Station structure, system, or component credited with mitigating any accident previously evaluated. The proposed change to modify the surveillance will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

The proposed change for LaSalle County Station, Units 1 and 2, implements Technical Specification Task Force Traveler 334 (TSTF–334), ''Relaxed Surveillance Frequency for Excess Flow Check Valve Testing,'' Revision 2. TSTF–334 notes that its implementation is only allowed for plants for which General Electric Nuclear Energy Topical Report NEDO–32977–A, ''Excess Flow Check Valve Testing Relaxation,'' is applicable. In addition, an EFCV performance criteria and basis must be developed to ensure that the corrective action program can provide meaningful feedback for appropriate corrective actions.

LaSalle County Station, in accordance with Topical Report NEDO–32977–A, has performed a plant-specific radiological dose assessment for an instrument line break, EFCV failure rate analysis, release frequency initiated by an instrument line break analysis and has proposed a corrective action program to ensure continued EFCV performance. The result of the assessment and analyses meets the overall requirements to allow implementation TSTF–334, Revision 2.

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 requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: June 28, 2002.

Description of amendment request: The proposed amendment would revise the Unit 1 Operating License and Technical Specifications to increase the licensed power level to 3304 megawatts thermal (MWt), or 1.66 percent greater than the current licensed power level of 3250 MWt.

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 of occurrence or consequences of an accident previously evaluated?

Response: No.

Probability of Occurrence of an Accident Previously Evaluated—In support of this measurement uncertainty recapture power uprate, a comprehensive evaluation was performed for nuclear steam supply system (NSSS) and balance of plant (BOP) components and analyses that could be affected by this change. A power calorimetric uncertainty calculation was performed, and the effect of increasing plant power by 1.66 percent on the plant's design and licensing basis was evaluated. The result of these evaluations is that all plant components will continue to be capable of performing their design function at an uprated core power of 3304 megawatts thermal (MWt). In addition, an evaluation of the accident analyses demonstrates that applicable analysis acceptance criteria continue to be met. No accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change.

Consequences of an Accident Previously Evaluated—This change does not affect the release paths, the frequency of release, or the source term for release for any accidents previously evaluated in the Updated Final Safety Analysis Report. Structures, systems, and components (SSC) required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3304 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition.

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.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. The installation of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system has been analyzed, and failures of this system will have no adverse effect on any safety-related system or any SSCs required for transient mitigation. SSCs previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. 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.

This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 3304 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3304 MWt. Credible malfunctions continue to be bounded by the current accident analysis of record or re-analysis demonstrates that applicable acceptance criteria continue to be met.

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

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

Response: No.

The margins of safety associated with this Measurement Uncertainty Recapture Uprate Program are those pertaining to core power. This includes those associated with the fuel cladding, Reactor Coolant System (RCS) pressure boundary, and containment barriers. A comprehensive engineering review was performed to evaluate the 1.66 percent increase in the licensed core power from 3250 MWt to 3304 MWt. The 1.66 percent increase required that revised NSSS design thermal and hydraulic parameters be established, which then served as the basis for all of the NSSS analyses and evaluations. This engineering review concluded that no design transient modifications are required to accommodate the revised NSSS design conditions. NSSS systems and components were evaluated and it was concluded that the NSSS equipment has sufficient margin to accommodate the 1.66 percent power uprate. NSSS accident analyses were either evaluated or revised for the 1.66 percent power uprate. In all cases the evaluations and re-analyses demonstrate that the applicable analyses acceptance criteria continue to be met. As such, the margins of safety continue to be bounded by the current analyses of record for this change.

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 requests involve no significant hazards consideration.

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

NRC Section Chief: L. Raghavan.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–220, Nine Mile Point Nuclear Station, Unit No. 1, Oswego County, New York

Date of amendment request: June 28, 2002.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) and, as applicable, other elements of the licensing bases to maintain a Post-Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," and Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to, or included in, the TSs for nuclear power reactors currently licensed to operate. However, lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means, or is of little use in the assessment and mitigation of accident conditions.

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949) on possible amendments to eliminate PASS, 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 March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the NSHC determination in its application dated June 28, 2002. The NSHC determination is restated below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC 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 PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

In the 20 years since the TMI–2 accident and the consequential promulgation of post accident sampling requirements, operating experience has demonstrated that a PASS provides little actual benefit to post accident mitigation. Past experience has indicated that there exists in-plant instrumentation and methodologies available in lieu of a PASS for collecting and assimilating information needed to assess core damage following an accident. Furthermore, the implementation of Severe Accident Management Guidance (SAMG) emphasizes accident management strategies based on in-plant instruments. These strategies provide guidance to the plant staff for mitigation and recovery from a severe accident. Based on current severe accident management strategies and guidelines, it is determined that the PASS provides little benefit to the plant staff in coping with an accident.

The regulatory requirements for the PASS can be eliminated without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. The elimination of the PASS will not prevent an accident management strategy that meets the initial intent of the post-TMI–2 accident guidance through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any 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 elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

Therefore, 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 [a] Margin of Safety

The elimination of the PASS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety. Methodologies that are not reliant on PASS are designed to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The use of a PASS is redundant and does not provide quick recognition of core events or rapid response to events in progress. The intent of the requirements established as a result of the TMI–2 accident can be adequately met without reliance on a PASS.

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.

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 7, 2002.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant Technical Specification (TS) Sections for administrative changes: (1) Section 1—''Definitions,'' (2) Section 2—''Safety Limits and Limiting Safety System Settings,'' (3) Section 5—''Design Features,'' and (4) Section 6—''Administrative Controls.'' The administrative changes include capitalizing defined words, formatting section titles, renumbering pages and correcting miscellaneous grammar and punctuation errors.

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

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

The proposed changes will not alter the intent of the TS. Reformatting the TS sections and correcting typographical, grammatical and format inconsistencies are administrative in nature. There is no impact on accident initiators or plant equipment, and therefore does not affect the probability or consequences of an accident.

2. 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 physical plant or operations. Since these are administrative changes they do not contribute to accident initiation. Therefore, the proposed changes do not produce a new accident scenario or produce a new type of equipment malfunction.

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

Since these are administrative changes, they do not involve a significant reduction in the margin of safety. The proposed changes do not affect plant equipment or operation. Safety limits and limiting safety system settings are not affected by this change.

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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: June 27, 2002.

Description of amendment request: SCE&G is proposing a revision to the Technical Specifications (TS) for the Virgil C. Summer Nuclear Station (VCSNS) to add an Allowed Outage Time (AOT) to Table 3.3–3, Engineered Safety Features Actuation System (ESFAS) instrumentation, Action Statement 16.

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:

South Carolina Electric & Gas Company (SCE&G) has evaluated the proposed changes to the VCSNS TS described above against the Significant Hazards Criteria of 10 CFR 50.92 and has determined that the changes do not involve any significant hazard. The following is provided in support of this conclusion.

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

The addition of an ACTION STATEMENT and the addition of an AOT (and its associated actions if not met) for a TS action statement are neither an accident initiator or precursor. The ESFAS actuates in response to an accident and has a mitigating function. Increasing the TS requirements for specific TS instrument loops provides additional assurance that the channels will be capable of performing their design function in the event of a DBA [design-basis accident]. The ability of the operations staff to respond to an evaluated accident or plant transient will not be hampered. This change provides conservative requirements to assure that the design basis of the plant is maintained.

Addition of conservative changes to the Engineered Safety Feature Actuation System Instrumentation [does] not contribute to the initiation of any accident evaluated in the FSAR [Final Safety Analysis Report]. Supporting factors are as follows:

—The changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–1431, Rev. 2.

There are no deletions from the Technical Specifications made by these changes, nor relaxation in any applicability, action, or surveillance requirements.

—Overall plant performance and operation [are] not altered by the proposed changes. There are to be no plant hardware changes as a result of this proposed change and only minimal procedural changes.

Therefore, since the Engineered Safety Feature Actuation System Instrumentation [is] treated more conservatively, the probability of occurrence or consequences of an accident evaluated in the VCSNS FSAR will be no greater than the original design basis of the plant.

Therefore, the change does 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?

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. Additionally, the addition of an ACTION STATEMENT and an AOT with conservative requirements are intended to assure that the plant is in a safe configuration and can meet accident analyses assumptions. These changes are conservative and consistent with the Improved Technical Specifications, NUREG–1431, Rev. 2. No new accident initiator mechanisms are introduced since:

—No physical changes to the Engineered Safety Feature Actuation System Instrumentation are made.

—No deletions from the Technical Specifications are made.

—No relaxation in any applicability, action, or surveillance requirements [is] made.

Since the safety and design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new accident scenarios have been created. Therefore, the types of accidents defined in the FSAR continue to represent the credible spectrum of events to be analyzed [that] determine safe plant operation.

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

The proposed change requires that an instrument channel for an Engineered Safety Feature [remains] operable or be restored to operability within a reasonable time period, otherwise a controlled shutdown is required. This conforms to the safety analysis where the plant and its systems, structures and components must be capable of performing the safety function while a DBA is occurring, in the presence of a worst case single failure.

This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

The proposed changes provide consistency between Tables 3.3–2, 3.3–3, and 4.3–2, resulting in a one-for-one correlation between the functional units in those tables. These changes are conservative and consistent with the Standard Technical Specifications, NUREG–0452, Rev. 5.

The proposed changes impose more restrictive operating limitations, and their use provides increased assurance that the Engineered Safety Feature Actuation System Instrumentation remains operable. Since the changes are conservative additions, it is concluded that the changes do not involve a significant reduction in the margin of safety. This is not a reduction in a margin of safety, since it restores the margin that was designed into the plant.

Pursuant to 10 CFR 50.91, the preceding analyses [provide] a determination that the proposed Technical Specifications change poses no significant hazard as delineated by 10 CFR 50.92.

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: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: May 8, 2002.

Description of amendment request: The proposed amendments would revise Technical Specifications (TS) Figure 2.1.1–1, ''Reactor Core Safety Limits;'' Table 3.3.1–1, ''Reactor Trip System Instrumentation;'' and the associated Bases B 2.1.1 and B 3.3.1.

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

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

The proposed change can be implemented without adverse impact to the safety analyses and plant systems. Implementation of the revised VEGP OTΔT [Overtemperature Delta Temperature] and OPΔT [Overpower Delta temperature] reactor trip setpoints will continue to ensure that fuel melt and departure from nucleate boiling (DNB) criteria are met. In addition, the setpoint changes will improve operating margin to the OTΔT and OPΔT reactor trip setpoints. The setpoints provide reactor protection and are not event initiators and therefore do not affect the probability of occurrence of an accident previously evaluated.

There is no change in the radiological consequences of any accident since the fuel clad, the reactor coolant system pressure boundary, and the containment are not changed, nor will the integrity of these physical barriers be challenged. In addition, the proposed change will not change, degrade, or prevent any reactor trip system actuations.

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?

The proposed change can be implemented without adverse impact to the safety analyses and plant systems. Implementation of the revised VEGP OTΔT and OPΔT reactor trip setpoints will continue to ensure that fuel melt and departure from nucleate boiling (DNB) criteria are met. In addition, the setpoint changes will improve operating margin to the OTΔT and OPΔT reactor trip setpoints. The revised OTΔT and OPΔT reactor trip setpoints would not create any new transients nor would they invalidate the OTΔT and OPΔT design bases.

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

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

The proposed change can be implemented without adverse impact to the safety analyses and plant systems. Implementation of the revised VEGP OTΔT and OPΔT reactor trip setpoints will continue to ensure that fuel melt and departure from nucleate boiling (DNB) criteria are met. In addition, the setpoint changes will improve operating margin to the OTΔT and OPΔT reactor trip setpoints. The margin of safety provided by the Technical Specifications is not significantly affected because the proposed changes are based on the same accident acceptance limits, i.e., the OTΔT and OPΔT design bases continue to be met.

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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: John A. Nakoski.

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

Date of application request: June 17, 2002. This application supercedes the December 6, 2001, application that was noticed in the Federal Register on February 5, 2002 (67 FR 5340).

Description of amendment request: The proposed amendment would revise the following Technical Specifications (TSs): (1) TS 3.3.6, ''Containment Purge Isolation Instrumentation;'' (2) TS 3.3.7, ''Control Room Emergency Ventilation System (CREVS) Instrumentation;'' (3) TS 3.3.8, ''Emergency Exhaust System (EES) Actuation Instrumentation;'' and (4) TS 3.9.4, ''Containment Penetrations.'' The revisions to the TSs affect limiting conditions for operation (LCOs), the required actions for LCOs, surveillance requirements, and tables specifying requirements on instrumentation. The revisions to the TSs are to allow the equipment hatch and the emergency air lock to be open in refueling outages during core alterations and/or movement of irradiated fuel within 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will allow the containment equipment hatch [and the emergency air lock] to be open during CORE ALTERATIONS and movement of irradiated fuel assemblies inside containment. The status of the containment equipment hatch or the emergency air lock during refueling operations has no [e]ffect on the probability of the occurrence of any accident previously evaluated. The proposed revision does not alter any plant equipment or operating practices in such a manner that the probability of an accident is increased. Since the consequences of a fuel handling accident inside containment with an open containment hatch [or emergency air lock] are bounded by the current analysis described in the FSAR [Final Safety Analysis Report] and the probability of an accident is not affected by the status of the containment equipment hatch [or emergency air lock], the proposed change[s] [do] not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes do not create any new failure modes for any system or component, nor do they adversely affect plant operation. No new equipment will be added and no new limiting single failures will be created. The plant will continue to be operated within the envelope of the existing safety analysis.

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

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

The previously determined radiological dose consequences for a fuel handling accident inside containment with the [equipment hatch or the] air lock doors open remain bounding for the proposed changes. These previously determined dose consequences were determined to be well within the limits of 10 CFR 100 and they meet the acceptance criteria of SRP [NRC Standard Review Plan] section 15.7.4 and GDC [NRC General Design Criterion] 19.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of application request: June 17, 2002.

Description of amendment request: The amendment would revise Technical Specification (TS) 3.3.1, ''Reactor Trip System (RTS) Instrumentation,'' by adding Surveillance Requirement (SR) 3.3.1.16 to Function 3 of TS Table 3.3.1–1. The amendment would add a requirement to verify the reactor trip system response times are within limits every 18 months on a staggered test basis.

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

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

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes.

The design of the RTS instrumentation, specifically the positive flux rate trip (PFRT) function, will be unaffected. The reactor protection system will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The proposed change imposes additional surveillance requirements to assure safety-related structures, systems, and components are verified to be consistent with the safety analysis and licensing basis. In this specific case, a response time verification requirement will be added to the PFRT function.

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

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR [final safety analysis report].

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

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

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance requirements. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

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

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

Therefore, the proposed 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.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed. The imposition of additional surveillance requirements increases the margin of safety by assuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

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

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 27, 2002. This application revises the application of September 27, 2001, that was originally noticed in the **Federal Register** on October 17, 2001 (66 FR 52805).

Description of amendment request: The proposed amendment would revise Section 5.3.1.1 of the Technical Specifications to state new education and experience eligibility requirements for operator license applicants.

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 TS change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program. [The change conforms] to the current requirements of 10 CFR 55.

Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. WCNOC's [Wolf Creek Nuclear Operating Corporation's] licensed operator training program is accredited by INPO [Institute for Nuclear Power Operations] and is based on a systems approach to training. The proposed TS change takes credit for the INPO accreditation of the licensed operator training program. The TS requirements for all other unit staff qualifications remain unchanged.

Therefore, the proposed change does not involve a signification 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 TS change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program and to conform to the revised 10 CFR 55.

As noted above, although licensed operator qualifications and training may have an indirect impact on the possibility of a new or different kind of accident from any accident previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised [10 CFR 55] rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and based on a systems approach to training. As previously noted, WCNOC's licensed operator training program is accredited by INPO and is based on a systems approach to training. The proposed TS change takes credit for the INPO accreditation of the licensed operator training program. The TS requirements for all other unit staff qualifications remain unchanged.

Additionally, the proposed TS change does not affect plant design, hardware, system operation, or procedures. 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.

The proposed TS change is an administrative change to clarify the current requirements applicable to licensed operator qualifications and licensed operator training program. This change is consistent with the requirements of 10 CFR 55. The TS qualification requirements for all other unit staff remain unchanged.

Licensed operator qualifications and training can have an indirect impact on a margin of safety. However, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 [rule], determined that this impact remains acceptable when licensees maintain a licensed operator training program that is accredited and based on a systems approach to training. As noted previously, WCNOC's licensed operator training program is accredited by INPO and is based on a systems approach to training.

The NRC has concluded, as stated in NUREG–1262, ''Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses,'' that the standards and guidelines applied by INPO in their training accreditation program are equivalent to those put forth or endorsed by the NRC. As a result, maintaining an INPO accredited, systems approach based licensed operator training program is equivalent to maintaining an NRC approved licensed operator training program which conform with applicable NRC Regulatory Guides or NRC endorsed industry standards. The margin of safety is maintained by virtue of maintaining an INPO accredited licensed operator training program.

In addition, the NRC has recently published NRC Regulatory Issue Summary 2001–01, ''Eligibility of Operator License Applicants,'' dated January 18, 2001, ''to familiarize addresses with the NRC's current guidelines for the qualification and training of reactor operator (RO) and senior operator (SO) license applicants.'' This document again acknowledges that the INPO National Academy for Nuclear Training (NANT) guidelines for education and experience, outline acceptable methods for implementing the NRC's regulations in this area.

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: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Stephen Dembek.

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/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.

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

Date of application for amendment: August 24, 2001, as supplemented June 11, 2002.

Brief description of amendment: The amendment revises the control room emergency filtration system requirements in Technical Specification 3.7.3, ''Control Room Emergency Filtration (CREF) System,'' based on NRC-approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Traveler TSTF–287, Revision 5, ''Ventilation System Envelope Allowed Outage Times.''

Date of issuance: June 28, 2002.

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

Amendment No.: 149.

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

Date of initial notice in ***Federal Register:*** May 28, 2002 (67 FR 36929). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 28, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 20, 2001.

Brief description of amendments: The amendments revised the Technical Specifications 5.6.5.b to eliminate the revision number and dates from the list of topical reports that contain the analytical methods used to determine the core operating limits.

Date of issuance: July 2, 2002.

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

Amendment Nos.: 199 and 192.

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

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10010). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 2, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 20, 2001.

Brief description of amendments: The amendments revised the Technical Specification 5.6.5.b to eliminate the revision number and dates from the list of topical reports that contain the analytical methods used to determine the core operating limits.

Date of issuance: July 10, 2002.

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

Amendment Nos.: 203 and 184.

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

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2921). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 10, 2002.

No significant hazards consideration comments received: No.

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: December 20, 2001.

Brief description of amendments: The amendments revised the Technical Specifications 5.6.5.b to eliminate the revision number and dates from the list of topical reports that contain the analytical methods used to determine the core operating limits.

Date of Issuance: July 9, 2002.

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

Amendment Nos.: 326, 326 and 327.

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

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10011).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 9, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 6, 2002, as supplemented by letter dated June 7, 2002.

Brief description of amendment: The amendment relocates the requirements for Main Steam Isolation Valve isolations on certain area temperatures from Technical Specification Section 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to the Technical Requirements Manual.

Date of issuance: July 11, 2002.

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

Amendment No.: 124.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 19, 2002 (67 FR 12601). The June 7, 2002, supplemental letter 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 July 11, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: March 31, 1999, as supplemented by letters dated June 1, July 14, and October 14, 1999, February 11, April 4 and 13, June 30, July 31, September 12 and 13, and October 23, 2000, May 31, October 18, 2001, and February 6, March 27, April 26, and June 11 and 12, 2002 (two letters).

Brief description of amendment: The amendment provides for the full conversion of the Current Technical Specifications to the Improved Technical Specifications.

Date of issuance: July 3, 2002.

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

Amendment No.: 274.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** November 8, 1999, (64 FR 60584), December 13, 1999, (64 FR 69574) and November 28, 2001 (66 FR 59595). The letters subsequent to the November 28, 2001, **Federal Register** notice did not change the technical content of the **Federal Register** notices, and did not change the scope of the proposed action. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 3, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 17, 2002.

Brief description of amendment: The amendment revises the KNPP Technical Specification (TS) 6.3, ''Plant Staff Qualifications,'' to change the title of the Superintendent Plant Radiation Protection to the Radiation Protection Manager. In addition, the licensee informed the Nuclear Regulatory Commission staff of its intention to reformat TS 6.3 using MicroSoft Word format.

Date of issuance: June 28, 2002.

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

Amendment No.: 161.

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

*Date of initial notice in **Federal Register:*** May 28, 2002 (67 FR 36932). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 28, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 7, 2001, as supplemented December 14, 2001 and April 1, 2002.

Brief description of amendment: Revised the Technical Specifications (TSs) to add a new condition and associated actions to Limiting Condition for Operation 3.8.1, ''AC Sources Operating,'' to allow one diesel generator to be out of service for 14 days.

Date of issuance: July 1, 2002.

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

Amendment No.: 39.

Facility Operating License No. NPF–90: Amendment revised the TSs.

*Date of initial notice in **Federal Register:*** September 19, 2001 (66 FR 48292). The supplemental letters provided clarifying information that was within the scope of the initial 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 July 1, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 12th of July, 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–18242 Filed 7–22–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Correction

The July 9, 2002, **Federal Register** contained a ''Biweekly Notice; Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.'' This notice corrects the notice published on July 9, 2002, (67 FR 45560). The last paragraph on page 45560 reads as follows: ''By July 25, 2002, the licensee may file a request for a hearing with * * *''. It should read, ''By August 8, 2002, the licensee may file a request for a hearing with * * *'' to correct the hearing date to 30 days.

Dated at Rockville, Maryland, this 17th day of July 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–18522 Filed 7–22–02; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension:

Rule 17f–5, SEC File No. 270–259,