informing 10 CFR 50, quality of PRAs, spent fuel pool fire safety study, more realistic (best estimate) thermal-hydraulic codes and status of ACRS activities on license renewals.

Friday, October 6, 2000

8:30 A.M.–8:35 A.M.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 A.M.–9:15 A.M.: Discussion of Topics for Meeting with the NRC Commissioners (Open)—The Committee will discuss matters scheduled for the meeting with the NRC Commissioners associated with risk informing 10 CFR 50 and related matters.

9:30 A.M.–12:00 Noon: Meeting with the NRC Commissioners (Open)—The Committee will meet with the NRC Commissioners, Commissioners' Conference Room, One White Flint North to discuss risk informing 10 CFR 50 and related matters.

1:30 P.M.–3:00 P.M.: Discussion of Industry Issues (Open)—The Committee will hear a presentation by R. Beedle, Senior Vice President, NEI on issues of mutual interest.

3:15 P.M.–4:45 P.M.: GSI–168, Equipment Qualification (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the GSI–168, Equipment Qualification.

4:45 P.M.–5:30 P.M.: ACRS Review of Generic Guidance Documents Associated with License Renewal (Open)—The Committee will discuss concerns identified during their initial review of the draft guidance documents.

5:30 P.M.–5:50 P.M.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

5:50 P.M.–6:00 P.M.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

6:00 P.M.–6:30 P.M.: Break and Preparation of Draft ACRS Reports (Open)—Cognizant ACRS members will prepare draft reports, as needed, for consideration by the full Committee.

6:30 P.M.–7:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, October 7, 2000

8:30 A.M.–8:35 A.M.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 A.M.–12:30 P.M.: Discussion of Proposed ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

12:30 P.M.–1 P.M.: Annual Report to the Commission on the NRC Safety Research Program (Open)—The Committee will discuss the format and content of the annual ACRS report to the Commission on the NRC Safety Research Program.

1 P.M.–1:30 P.M.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 28, 1999 (64 FR 52353). In accordance with these procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting and questions may be asked only by members of the Committee, its consultants, and staff. Persons desiring to make oral statements should notify Mr. James E. Lyons, ACRS, five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting Mr. James E. Lyons prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with Mr. James E. Lyons if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Mr. James E. Lyons (telephone 301–415–7371), between 7:30 a.m. and 4:15 p.m., EDT.

ACRS meeting agenda, meeting transcripts, and letter reports are available for downloading or viewing on the internet at http://www.nrc.gov/ACRSACNW.

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., EDT, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: September 14, 2000.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 00–24160 Filed 9–19–00; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 28, 2000, through September 8, 2000. The last biweekly notice was published on September 6, 2000 (65 FR 54083).

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 Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC through September 22, 2000. The NRC is relocating its Public Document Room to the NRC's headquarters building. Effective September 26, 2000, documents may be examined at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By October 20, 2000, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC through September 22, 2000 or at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852 effective September 26, 2000, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC through September 22, 2000 or at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852 effective September 26, 2000, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC through September 22, 2000 or at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852 effective September 26, 2000, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Carolina Power & Light Company, et al., Docket Nos. 50–325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: July 27, 2000.

Description of amendments request: The proposed license amendments would revise Technical Specification (TS) 3.8.5, ''DC Sources—Shutdown.'' The operability requirements for the DC sources, during shutdown conditions, would be revised to require one of the unit's DC electrical power subsystems to be operable when in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary 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:

In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. Revising the operability requirements for the DC sources, during shutdown conditions, to require one of the unit's DC electrical power subsystem to be operable when in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operability of the DC electrical power sources during Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

As stated in TSTF–204, Revision 3, worst case design basis accidents which are analyzed for operating modes are not as significant of a concern during shutdown modes due to lower energy levels. The TSs, therefore, require a lesser complement of electrical equipment to be available during shutdown than is required during operating modes. Specifically, assuming a single failure concurrent with a loss of all offsite or all onsite power is not required. This concept is consistent with the BSEP TSs, prior to conversion to ITS [Improved TS], in that TS 3.8.2.4.2 required either Division I or Division II of the DC power distribution system to be operable when in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment. The operability requirements of the DC electrical power sources for a unit in Modes 1, 2, and 3 are not affected by the proposed amendments.

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

2. Revising the operability requirements for the DC sources, during shutdown conditions, to require one of the unit's DC electrical power subsystem to be operable when in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Revising the operability requirements of TS 3.8.5 does not involve physical modification to the plant and does not introduce a new mode of operation. Therefore, there is no possibility of an accident of a new or different type.

3. Revising the operability requirements for the DC sources, during shutdown conditions, to require one of the unit's DC electrical power subsystem to be operable when in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment does not involve a significant reduction in a margin of safety.

The proposed change revises LCO [Limiting Condition for Operation] 3.8.5 to require one of the unit's DC electrical power subsystems to be operable when the unit is in Modes 4 and 5 and during movement of irradiated fuel assemblies in the secondary containment. This is acceptable due to the lower energy levels involved with potential accidents occurring during shutdown modes and because assuming a single failure concurrent with a loss of all offsite or all onsite power during such events is not required. This is consistent with the TS requirements, as they existed prior to conversion to ITS and TSTF–204, Revision 3 which was approved by the NRC on February 16, 2000. The operability requirements of the DC electrical power sources for a unit in Modes 1, 2, and 3 are not affected by the proposed amendments.

Based on the above, the proposed amendments 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Richard P. Correia.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: August 22, 2000.

Description of amendment request: The proposed amendment would change: (1) Technical Specification (TS) 3.10.4, ''Rod Insertion Limits,'' to allow on-line calibration of the rod position indicator (RPI) channels during operating cycle 15, and (2) TS 3.10.6, ''Inoperable Rod Position Indicator Channels,'' to allow extended RPI deviation limits during cycle 15.

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 license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

No. Neither the probability not the consequences of an accident previously analyzed is increased due to the proposed changes. All peaking factors will remain within the limits of the Technical Specifications. Both the shutdown margin and the axial flux difference will be maintained within the limits of the Technical Specifications. There will be no fuel damage due to the changes. All design and safety criteria will be met. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

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

No. The changes will not create the possibility of a new or different kind of accident. The calibration will be performed using plant procedures that have been reviewed and approved by Con Edison's Station Nuclear Safety Committee (SNSC). It has been shown that even with the new RPI deviation bands and on-line calibration, all power distribution limits will be met. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment does not involve a significnat reduction in the margin of safety. There will be no change in the power distribution limits used in the design and safety analyses and the required shutdown margin will be maintained. It has been shown that there is no fuel failure as a result of this change. 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: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Marsha Gamberoni.

Consumers Energy Company, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: August 28, 2000.

Description of amendment request: The proposed amendment would revise the date for implementation of the Palisades Plant Improved Technical Specifications (ITS) from on or before October 31, 2000, to on or before December 31, 2000. The current implementation date was established by previous Amendment No. 189, dated November 30, 1999, in which the NRC staff stated that Amendment No. 189 was ''effective as of its date of issuance and shall be implemented on or before October 31, 2000.'' The proposed amendment would change this date to December 31, 2000.

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

A discussion of these [10 CFR 50.92] standards as they relate to this request follows to show that operation of the facility in accordance with the proposed change does not:

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

The proposed change is administrative in nature in that it simply extends the date for implementation of the Improved Technical Specifications (ITS) from October 31, 2000 to December 31, 2000. The proposed extension of the ITS implementation date is necessary in order to allow for additional Operations shift crew training and readiness assessment, as well as a longer transition period of operating the plant using the Current Technical Specifications (CTS) and ITS in parallel. These actions are considered essential to proper ITS implementation. Until ITS are implemented, the previously approved CTS will remain in effect and the unit will continue to be operated in accordance with the NRC approved CTS requirements.

The proposed change is administrative in nature, and does not involve any changes to the design or operation of the Palisades Plant which may affect the probability or consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). Previously evaluated accident precursors or initiators are not affected and, as a result, the probability of accident initiation will remain as previously evaluated. There will be no impact on the capability of any structures, systems or components to perform their credited safety functions to prevent an accident or mitigate the consequences of an accident previously evaluated. Therefore, the probability or consequences of a postulated accident previously evaluated in the UFSAR are not increased as a result of the proposed change.

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

Extension of the date for ITS implementation is an administrative change. The proposed change does not involve any changes to the physical structures, components, or systems of the Palisades Plant. Since the change is administrative, there will be no impact on the process variables, characteristics, or functional performance of any structures, systems or components in a manner that could create a new failure mode. Further, the change will not introduce any new modes of plant operation or eliminate any actions required to prevent or mitigate accidents. Thus, operation in accordance with the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Extension of the date for ITS implementation is an administrative change. The proposed change does not involve any hardware changes or physical alteration of the plant and the change will have no impact on the design, design basis, or operation of the plant. The change will not eliminate any requirements, impose any new requirements, or alter any physical parameters which could reduce any margin of safety. The continued operation of Palisades in accordance with the previously approved Current Technical Specifications assures the proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Claudia M. Craig.

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: July 18, 2000, supplemented by letter dated August 22, 2000.

Description of amendment request: The proposed amendments would move Technical Specifications Sections 3.3.11, 3.3.12, and 3.3.13 (and the corresponding Bases) that specify the Main Steam Line Break requirements by renumbering them 3.3.25, 3.3.26, and 3.3.27 and indicating that these requirements will remain in effect for each unit until after the Automatic Feedwater Isolation System (AFIS) is installed on the unit. In addition, the proposed amendments would incorporate requirements and Bases for a new AFIS that will become effective when the modification is installed on each unit. These requirements will become Sections 3.3.11, 3.3.12, and 3.3.13.

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

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

No. There is no significant increase in the probability of a loss of Main Feedwater (MFW) or Emergency Feedwater (EFW) event due to spurious actuation of the Automatic Feedwater Isolation System (AFIS). AFIS provides a means of automatic response to improve the ability to isolate MFW and EFW to mitigate containment overpressurization and steam generator tube stresses resulting from Main Steam Line Break (MSLB) accidents. AFIS replaces the need for manual operator actions currently required by emergency operating procedures, but these remain as a defense-in-depth. AFIS is highly reliable, being designed with two independent trains of diverse digital control systems, each having four channels of inputs. The AFIS modification will also upgrade some existing components that were actuated by MSLB Detection and Feedwater (FDW) Isolation Circuitry that were not safety grade to safety grade quality thereby improving system reliability. Therefore, the installation of AFIS does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

No. AFIS replaces the MSLB Detection and FDW Isolation Circuitry as described in the Updated Final Safety Analysis Report. AFIS is capable of determining which steam generator has been affected and will isolate MFW and EFW to that steam generator. In this regard, AFIS performs the same functions that are currently performed by the combination of the MSLB Detection and FDW Isolation Circuitry plus the additional operator actions needed to isolate the affected steam generator. Safety features have been designed into AFIS to prevent spurious actuation. The system must be energized to trip therefore AFIS will not cause a trip on loss of power. There are no postulated failures such as loss of power that differ from those assumed for an analog control system that would prevent proper system actuation. The design of the two out of four input logic provides redundancy against the affects of single failures that could cause spurious actuation. In the unlikely event of spurious actuation, manual manipulation of EFW pump controls will override the AFIS trip signals. Therefore, AFIS does not introduce hardware failures that inhibit proper operation of MFW or EFW. In conclusion, AFIS does not create the possibility of a new or different kind of accident from any kind previously evaluated.

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

No. The proposed change does not adversely affect any plant safety limits, setpoints, or design parameters. The change also does not adversely affect the fuel, fuel cladding, Reactor Coolant System, or containment integrity. For a postulated feedwater line break (FLB)/MSLB inside containment, AFIS will improve the margin of safety by reducing the mass and energy release to containment. AFIS will also improve the margin of safety for departure from nucleate boiling by minimizing the overcooling transient from a FLB/MSLB. 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: August 10, 2000.

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.9.4, "Refueling Operations, Containment Building Penetrations," by deleting the requirements for the containment purge and exhaust system and by revising the closure requirements for containment building penetrations to require that containment penetrations are capable of being closed during the handling of irradiated fuel within the 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:

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

The containment purge and exhaust system is not considered an accident initiator nor do the proposed changes result in any physical change to the plant design. Therefore the probability of an accident previously analyzed remains unchanged. In addition, the containment purge and exhaust system filtration units are not credited in the ANO–2 [Arkansas Nuclear One, Unit 2] safety analysis in limiting offsite dose consequences during an accident. Furthermore, the system is designed to automatically isolate, as required by ANO–2 TS [Technical Specification] 3.3.3.1, Table 3.3–6 upon receipt of a high radiation signal when in operation in Modes 5 and 6. Since the containment purge and exhaust system is credited only for long-term post accident cleanup efforts and will continue to be tested to ensure the filtration system remains effective in supporting such efforts, the consequences of an accident previously evaluated remains unchanged.

The opening of a containment penetration during the handling of irradiated fuel within the containment building is limited to Mode 6 with the core flooded to refueling level (≥ 23 feet of borated water above the fuel) by the applicability of TS 3.9.4. Such openings are strictly controlled by safe shutdown programs such as the ANO–2 Shutdown Operations Protection Plan (SOPP) and the Outage Risk Management Guidelines (ORMG). A containment penetration being open during the handling of irradiated fuel does not result in an increase in the probability of an accident that has been previously evaluated.

ANO [Arkansas Nuclear One] submitted the radiological dose consequences of a fuel handling accident within the containment building to the NRC [Nuclear Regulatory Commission], illustrating that without a containment building, the offsite dose consequences due to a fuel handling accident inside containment would remain well within 10 CFR 100 limits. This evaluation was approved by the NRC in Amendment 166 to the ANO–2 Operating License and referenced in the aforementioned Amendment 203 to the ANO–2 Operating License in support of allowing the equipment hatch and/or personnel air locks to remain open during fuel handling activities. Since the above evaluation assumes no credit for "containment" and subsequently illustrates that the resulting offsite dose consequences are acceptable, the consequences of an accident previously evaluated are not adversely impacted.

The proposed revision of penetration closure methods does not impact any accident previously analyzed or impact the consequences of such an accident. The licensee will continue to be accountable for ensuring adequate and timely closure of each containment penetration should such closure become necessary. Revising the examples given in the TSs for establishing closure is, therefore, considered risk-neutral and is consistent with the Revised Standard Technical Specifications (RSTS) of NUREG–1432.

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

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

The containment purge and exhaust system filtration units are not credited in the ANO–2 safety analysis in limiting offsite dose consequences during an accident. Furthermore, the system is designed to automatically isolate, as required by ANO–2 TS 3.3.3.1, Table 3.3–6, upon receipt of a high radiation signal when in operation in Modes 5 and 6. Since the containment purge and exhaust system is credited only for long-term post accident cleanup efforts and will continue to be tested to ensure the filtration system remains effective in supporting such efforts, the possibility of a new or different kind of accident being created from that previously evaluated remains unchanged.

The fuel handling accident has previously been addressed in the ANO–2 safety analysis. In addition, the offsite dose consequences of the fuel handling accident have been found to be acceptable while assuming no credit for containment. Therefore, the provision to allow penetrations to be opened during the handling of irradiated fuel within the containment building does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed revision of penetration closure methods is also not considered an accident initiator. As an added measure of safety, however, the appropriate administrative controls required by Amendment 203 to the ANO–2 Operating License will be applicable to the containment penetrations impacted by the relevant proposals of this submittal.

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

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

The containment purge and exhaust system is not presently permitted to be placed in operation in Modes 1, 2, 3, or 4 and thus eliminates one possible path for radiological release to the public. The automatic actuations discussed above that act to isolate the system during emergency events in Modes 5 and 6 also provide assurance that a radiological release will not occur via the containment purge and exhaust system flow paths. Furthermore, the containment purge and exhaust system filtration units are not credited in the ANO–2 safety analysis in limiting offsite dose consequences during an accident. Since the containment purge and exhaust system is credited only for long-term post accident cleanup efforts and will continue to be tested to ensure the filtration system remains effective in supporting such efforts, the margin to safety remains unchanged.

ANO–2 has provided sufficient information to illustrate that the offsite dose consequences, as a result of a fuel handling accident, remain well within 10 CFR [Part] 100 limits, while assuming no credit for containment for release mitigation. Since no increase in the offsite dose potential is evident due to the opening of containment penetrations, the margin to safety is not adversely affected by this proposed revision.

The proposed revision of penetration closure methods does not impact the margin to safety. The licensee will continue to be accountable for ensuring adequate and timely closure of each containment penetration should such closure become necessary.

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

NRC Section Chief: Robert A. Gramm.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Dade County, Florida

Date of amendment request: August 18, 2000.

Description of amendment request: The proposed amendments would revise the current requirements of Technical Specifications (TS) 3.3.2, Engineered Safety Features Actuation System Instrumentation, Table 3.3–2, Items 7.b and 7.c. Specifically, the proposed license amendments revise ACTION statement 18 to allow operation of the units with both channels of undervoltage protection bypassed for up to 8 hours to allow performance of the monthly surveillance without placing the units in a condition prohibited by the TS. In addition, an administrative change to Item 7.b. of TS Tables 3.3–2, 3.3–3, and 4.3–2 is requested to change "Degraded Voltage" to "Undervoltage" to make it consistent with the Updated Final Safety Analysis Report description.

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

Approval and implementation of this amendment will have no effect on the probability or consequences of accident previously evaluated. The proposed changes allow performance of the required surveillance without placing the plant in a condition prohibited by the Technical Specifications. The undervoltage and degraded voltage protection schemes of the 480 volt load centers are not affected. Therefore, there will be no impact on any accident probabilities by the approval of this amendment. Therefore, the proposed amendments do not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not alter the design, physical configuration, or modes of operation of the plant. No changes are being made to the plant that would introduce any new accident causal mechanisms. The proposed Technical Specification changes do not impact any plant systems that are accident initiators, since the 480 volt undervoltage and degraded voltage protection logics are not affected. The proposed change allows performance of the required surveillance without placing the plant in a condition prohibited by the Technical Specifications. No new accident causal mechanisms are created as a result of NRC approval of the proposed amendments request. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes do not change the operation, function or modes of plant or equipment operation. The proposed changes do not change the undervoltage and degraded voltage protection logics of the 480 volt load centers. The ability of the 480 volt load center voltage protection schemes to detect degraded voltage and initiate a signal to the sequencers is maintained. No new hazards or failure modes are created or postulated which may cause an accident different from any accident previously analyzed. The proposed changes revise ACTION statement 18 to allow performance of the technical specification required surveillances without placing the plant in a condition prohibited by the Technical Specifications. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment requests: August 18, 2000.

Description of amendment requests: The proposed amendments would change Technical Specification (TS) 3/4.7.4, "Essential Service Water [ESW] System," and the associated Bases to add requirements that would support cross-connection to the opposite unit. The proposed amendment would also delete a provision for a 60-day allowed outage time when an ESW flowpath is not available to support the opposite unit's shutdown functions. Administrative and editorial changes are also made to provide consistency between units, correct typographical errors, improve readability, and improve page layout.

The licensee is submitting this request in accordance with Nuclear Regulatory Commission (NRC) Administrative Letter 98–10, ''Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety,'' because the current TS requirements are nonconservative. The licensee determined that an operable ESW pump may be adversely affected by inoperability of an opposite unit ESW pump sharing the same header. With open crosstie valves on the header, an inoperable pump can permit flow to be diverted from the operable ESW pump to the loads on the opposite unit. This could be safety significant when the operable pump is supplying accident loads.

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

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

The accidents previously evaluated in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) that are affected by operation of the essential service water system are:

1. Loss-of-Coolant Accident (LOCA),
2. Main Steam Line Break,
3. Feedwater Line Break,
4. Loss of Feedwater,
5. Combinations of the above accidents with loss of offsite power,
6. Appendix R fire, and
7. Flooding.

The closing of an ESW unit crosstie valve to isolate an operating ESW pump from an inoperable loop will not increase the probability of occurrence of the affected accidents. Closing these valves will not affect the initiators of any previously analyzed accidents. It prevents flow in an operating loop from being reduced below design basis by flow diversion to an inoperable loop. This action will not affect the initiating frequency of any LOCAs, main steam line or feedwater line breaks, or loss of feedwater events, nor will it cause or increase the frequency of an Appendix R fire.

With respect to flooding, closing the ESW unit crosstie valves may reduce the extent of flooding should a break occur in the ESW system. It does not contribute to the probability of an ESW system pipe break occurring. Therefore, closing the unit crosstie valve(s) as directed by the revised T/S requirements is a conservative change relative to flooding.

Closing an ESW unit crosstie valve to isolate an operating ESW pump from an inoperable loop will not increase the consequences of any accident previously evaluated in the [Updated Final Safety Analysis Report] UFSAR. This configuration does not prevent the ESW system from meeting its design basis flow requirements because these flows are set with the crosstie valves closed.

As long as the ESW design basis flow requirements are met, this proposed change is bounded by the current analysis of record with respect to consequences. No new release paths are created and the frequency of release is not increased by closing the unit crosstie valve when required by the revised requirements. Preventing the diversion of flow from an operating to an inoperable loop will reduce the probability of equipment malfunction that could lead to an increase in the consequences of an accident. Loss of offsite power in conjunction with any of the affected accidents will not be impacted by closure of the crosstie valve(s) because the valves receive emergency power.

The change to delete the additional 60-day allowed outage time (AOT) of the shutdown flowpath to the opposite unit is a conservative change that only increases the availability of the shutdown flowpath.

The change to add T/S 4.0.5 to the Unit 2 surveillance is a conservative change that corrects an editorial oversight. Surveillance testing under T/S 4.0.5 has been previously evaluated and approved.

The remaining changes are administrative in nature and are not intended to change the meaning of the T/S or associated Bases.

Therefore, these changes cannot increase the consequences or probability of occurrence 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?

Closing an ESW unit crosstie valve to isolate an operating ESW pump from an inoperable loop will not create the possibility of an accident of a new or different type than any previously evaluated. Operation with closed crosstie valves is not the normal operating lineup but it is not precluded and applicable procedures recognize they may be closed. Therefore, no system/component interfaces are affected, nor are new ones created that would contribute toward a new or different accident. As described in question 1 above, operation with closed crosstie valves is bounded by the current analysis for affected accidents, even if these are combined with a loss of offsite power. Other single failures in conjunction with this change, such as the loss of one train of emergency diesel generators on one unit, will not create an accident that is not bounded by the current analysis of record.

The change to delete the additional 60-day AOT of the shutdown flowpath to the opposite unit is a conservative change that only increases the availability of the shutdown flowpath.

The change to add T/S 4.0.5 to the Unit 2 surveillance is a conservative change that corrects an editorial oversight. Surveillance testing under T/S 4.0.5 has been previously evaluated and approved and is included in the Unit 1 surveillance requirements.

The remaining changes are administrative/ editorial in nature and are not intended to change the meaning of the T/S or associated Bases.

Therefore, this proposed change does not increase the possibility of a new or different kind of accident than previously evaluated.

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

Closing an ESW unit crosstie valve to isolate an operating ESW pump from an inoperable loop ensures the single-failure design of the ESW system will be maintained. In this manner, the system will continue to perform its required function and ensure that margins of safety is maintained.

The change to delete the additional 60-day AOT of the shutdown flowpath to the opposite unit is a conservative change that assures the availability of the shutdown flowpath.

The change to add T/S 4.0.5 to the Unit 2 surveillance is a conservative change that corrects an editorial oversight. Surveillance testing under T/S 4.0.5 has been previously evaluated and approved and is included in the Unit 1 surveillance requirements.

The remaining changes are administrative in nature and are not intended to change the meaning of the T/S or associated Bases.

Therefore, these 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 requests involve no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: September 1, 2000.

Description of amendment requests: The proposed amendments would clarify Technical Specification (TS) 3/4.4.4, ''Pressurizer,'' to reflect the current power supply to the pressurizer heaters and require two operable trains of pressurizer heaters during Modes 1, 2, and 3. The proposed amendments also revise the Bases for TS 3/4.4.4 to reflect these changes and to clarify the purpose of the pressurizer heaters.

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:

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

The proposed changes do not affect any accident initiators or precursors. Neither the pressurizer heaters nor the surveillance test is an accident initiator. Therefore, the proposed changes will not affect the probability of an accident.

The pressurizer heaters are not credited to mitigate the consequences of any accidents evaluated in the Updated Final Safety Analysis Report; however, they are needed during a loss of offsite power to provide adequate subcooling margin in the reactor coolant system so that natural circulation conditions can be maintained at hot standby. The proposed change to reflect the current power supply to the pressurizer heaters modifies the surveillance requirement to reflect the design and eliminates redundancy with other surveillance requirements. Components will continue to be tested just as frequently. The proposed change to require two trains of heaters instead of one will provide better assurance that the required capacity is operable.

Overall, testing the same components and requiring redundant capacity provides assurance that the required function will be performed as assumed. As such, the consequences of an accident will not significantly increase.

Therefore, the probability of occurrence or consequences of accidents previously evaluated are not increased.

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

The proposed changes do not modify any equipment or the operational limits of any equipment. The proposed changes do not introduce any new failure mechanisms to the pressurizer or any other plant systems. The proposed changes do not change the method by which any plant system performs its function.

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

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

T/S 3.4.4 requires pressurizer heater capacity of at least 150 kW to provide adequate subcooling margin in the reactor coolant system during a loss of offsite power condition to maintain natural circulation conditions at hot standby. The proposed changes will increase the requirements for pressurizer heaters by requiring two trains of pressurizer heaters to be operable with at least 150 kW in each train instead of 150 kW total capacity. This provides assurance that the required function will be performed as assumed.

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 requests involve no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: September 1, 2000.

Description of amendment requests: The proposed amendments would change Technical Specification (TS) surveillance requirement 4.6.1.2 and the associated T/S Bases to address exemptions to leakage rate testing specified by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' and Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' dated September 1995.

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

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

The proposed in-service testing does not affect accident initiators or precursors. The proposed ASME Section XI in-service test is routinely performed to collect data while the plant is in Mode 3. Conducting the containment leakage test in Mode 3 rather than prior to Mode 4 entry does not affect the probability of an accident. Excessive containment leakage is not a factor until after an accident has already occurred.

The proposed in-service testing does not affect the containment leakage rate limits. Therefore, the consequences of an accident are unchanged. The proposed change to conduct the testing in Mode 3 would not significantly increase the consequences of an accident. In order to have a release through the modified closed piping systems, there would need to be a loss-of-coolant accident concurrent with a through-wall leak, with enough pressure in containment to overcome main steam system pressure. These conditions are extremely unlikely to occur simultaneously in Modes 3 and 4.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

The proposed changes do not introduce any additional physical interface with plant equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. The proposed changes demonstrate the leak-tight integrity of the affected portions of the containment barrier through the performance of a visual inspection for through-wall leakage.

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

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

T/S 3.6.1.2 is based on limiting total containment leakage volume to the value assumed in the accident analysis at peak accident pressure. The proposed change does not change the allowable leakage rates.

Since the in-service test is performed at a significantly higher pressure than the Type A test and the in-service test acceptance criterion is zero through-wall leakage, versus a nominal amount allowed for the Type A test, the margin of safety will not be reduced. The proposed change would demonstrate the leak-tight integrity of the steam generator and associated piping, as components of the containment barrier, in a fashion at least as rigorous as the Type A test. If the leakage from containment is maintained within the T/S limit, dose rates at the site boundary will not be increased.

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

In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no 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 requests involve no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of amendment request: September 1, 2000.

Description of amendment request: The proposed amendment would resolve an unreviewed safety question dealing with the licensee revising the Unit 1 and 2 safety analyses to incorporate changes regarding modeling of pressurizer heater operation and spray effectiveness as they relate to certain transients that are analyzed for pressurizer overfill. As a part of the revision to the Unit 2 analyses only, the licensee proposes to change the value of the moderator temperature coefficient (MTC) assumed as an initial condition for these transients. The licensee evaluated the proposed change to the MTC value pursuant to 10 CFR 50.59 and determined that the proposed change constituted an unreviewed safety question. Therefore, in accordance with 10 CFR 50.90 the licensee is seeking approval on its use of a different value for MTC as an input assumption for analyses of these transients on Unit 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:

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

The proposed change (*i.e.*, revise MTC assumption) and changes already implemented (*i.e.*, revised modeling of pressurizer heaters and sprays) result in more conservative modeling of transient analyses and do not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The proposed change and changes already implemented would affect the analyzed reactor coolant system (RCS) response to a [loss of all non-emergency alternating current] LOAC or [loss of normal feedwater] LONF transient. Operational occurrences that are postulated to occur on a moderate frequency, such as the LOAC and LONF transients, are analyzed to ensure they do not generate a more serious plant condition without other faults occurring independently. Specifically, these events are analyzed to ensure that pressurizer overfill would not occur. If pressurizer overfill occurs, liquid could pass through the power-operated relief valves or the safety valves. Since these valves are qualified to pass steam and not liquid, there is a potential to fail one of these valves in the open position, creating an uncontrolled release of primary coolant. An uncontrolled release of primary coolant through a failed-open relief or safety valve would be considered a small break loss-of-coolant accident (SBLOCA). Because a loss-of-coolant accident is a more serious condition than a LOAC or LONF transient, this would constitute a violation of the acceptance criterion discussed above. The changes already implemented affect the approach to modeling the pressurizer heaters and sprays in the accident analysis and the proposed Unit 2 MTC change affects an input assumption to the analyses, but none of these changes result in a revision to the acceptance criteria for a change in the probability of occurrence of these events.

The changes to the pressurizer heater and spray modeling assumptions result in a more conservative outcome for the LOAC and LONF transients. When considered in concert with the proposed change to reduce the assumed MTC value, the revised LOAC and LONF analyses yield acceptable results (pressurizer overfill does not occur). Pressurizer heaters and sprays are control systems that are used to modulate the primary coolant pressure during normal operation, and during certain postulated accident scenarios. Neither of these control systems are considered precursors or initiators to any accidents described in the Updated Final Safety Analysis Report (UFSAR). The change to conservatively assume the heaters are in operation during a LOAC or LONF transient does not affect the actual design or operation of the heaters during any mode of operation. Similarly, revising the assumed effectiveness of the sprays in the LOAC and LONF accident analyses has no effect on the actual design or operation of the sprays. Consequently, the changes to the assumed operation of the sprays and heaters in the LOAC and LONF accident analyses would not cause either of these control systems to become an initiator or precursor of an accident. The MTC is an analysis input that affects the way the plant responds during a temperature transient. Reducing the MTC assumed in the analysis to a more restrictive value will result in a less severe response of the reactor core and RCS to the LOAC and LONF transients. The MTC assumed for a safety analysis does not initiate any accident scenarios. Changing the MTC as an assumed input to the analysis does not result in an increase in the frequency of any initiating events. Therefore, these changes do not increase the probability of a previously evaluated accident.

The operation of pressurizer heaters and sprays has no direct impact on radiological consequences of a LOAC, LONF, or any other previously analyzed event. Similarly, the assumed MTC does not directly impact the source term or radiological release pathways for any previously analyzed events. Revising the LOAC and LONF analyses to conservatively model the pressurizer heaters and sprays and to assume a more restrictive MTC value results in precluding the occurrence of a pressurizer overfill condition. Consequently, the revised LOAC and LONF analyses demonstrate that these transients would not progress to the occurrence of a SBLOCA. Therefore, these analytical changes do not result in an increase in the consequences for these transients.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The changes in modeling and assumptions for the LOAC and LONF transients do not create the possibility of a new or different kind of accident from any accident previously evaluated. The changes to the modeling of the pressurizer heaters and sprays are analysis assumption changes that result in a more conservative outcome for the LOAC and LONF transients.

Because the changes do not alter the design or operation of the pressurizer heaters or sprays, they do not introduce any new malfunctions. The changes pertain to the correction of analysis assumptions for modeling the pressurizer control features for the two UFSAR events that have been evaluated. The only potential outcome of the application of the heater and revised spray models causing the analyses to exceed the acceptance criteria is another event (*i.e.,* SBLOCA) that has been evaluated in the UFSAR. Consequently, the changes to the modeling of pressurizer heaters and sprays in the LOAC and LONF transients cannot affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

The MTC is an analysis input that affects the way a plant responds during a temperature transient. Reducing the MTC assumed in the analysis to a more restrictive value will result in a less severe response of the reactor core and RCS to the LOAC and LONF transients. As used in these analyses, the assumed MTC value is not a factor in initiating any accident scenarios. Consequently, the application of a lower MTC value to the analyses of the LOAC and LONF transients cannot affect or create new accident initiators or precursors or create the possibility of a new or different kind of accident.

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

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

The changes to modeling and assumptions for the LOAC and LONF transients do not involve a significant reduction in a margin of safety. The changes to the modeling of the pressurizer heaters and sprays result in a more conservative outcome for the LOAC and LONF transients. The acceptance criterion for events analyzed for pressurizer overfill, such as the LOAC and LONF transients, is that the pressurizer does not reach a water-solid condition. In order for the Unit 2 analyses to meet this acceptance criterion, it was necessary to change the assumed MTC from a positive value to zero at full-power conditions. However, a positive MTC has been assumed in previous NRC analyses of these transients in support of past [Donald C. Cook Nuclear Plant] CNP license amendments. Specifically, the NRC's approval of the current Unit 2 Technical Specification (T/S) 3/4.1.1.4 MTC curve (License Amendment 107 to DPR–74) was predicated, in part, on the basis that the safety analysis assumptions remain valid. Because a positive MTC was assumed in previous safety analyses, and the proposed MTC value is less conservative than the positive value assumed in those previous safety analyses, this activity is a reduction in the margin of safety.

T/S 3/4.1.1.4, Figure 3.1–2, specifies the operational limits for the MTC. The T/S allows a constant MTC of +5 pcm/°F for core power levels from 0% to 70%. Above 70% power, the allowed MTC value ramps down linearly to 0 pcm/°F at full power. The basis for the limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the UFSAR accident and transient analyses. Although the revised initial MTC value assumed in these analyses is reduced from that assumed in the current analyses of record in the CNP UFSAR, it is still within the requirements of Unit 2 T/S 3.1.1.4 for 100% power. Thus, the revised MTC value remains bounding for full-power operation. Furthermore, the analyses for the LONF and LOAC scenarios both assume an initial reactor power of 102%, which also bounds full power operation. Consequently, the revised assumption for the Unit 2 MTC ensures that the conditions assumed for the transient evaluation still bound the most limiting plant operating conditions and are consistent with the requirements in T/S 3.1.1.4. Thus, the basis for approval of the current Unit 2 MTC curve, as specified by the safety evaluation report that approved this curve (Amendment No. 107 to DPR–74), remains valid.

A sensitivity study was performed to confirm that the basis for establishing an MTC of 0 pcm/°F at full power bounds partial-power conditions with the corresponding positive MTC. Specific LOAC and LONF calculations were performed by varying (reducing) the nominal core power levels and assuming the corresponding positive MTC values at each core power level. The result of the study confirmed that, for both the LOAC and LONF events, the full power case with an MTC value of 0 pcm/°F bounds the case with a positive MTC initialized at a lower power level.

By revising the assumed MTC value from a positive value to zero, the Unit 2 LOAC and LONF analyses demonstrate that the analysis acceptance criteria are met (*i.e.,* pressurizer overfill does not occur) and bound the positive MTC cases at lower power levels. Therefore, the combination of these three analytical modeling changes results in an acceptable analytical outcome for Unit 2. Furthermore, the zero MTC value is still within the requirements of Unit 2 T/S 3.1.1.4 for 100% power. Thus, the revised MTC value remains bounding for full-power operation. Although the proposed MTC assumed in the LOAC and LONF analyses are a reduction in the margin provided to the NRC in previous evaluations of these transients, the use of the full-power MTC is consistent with the plant T/S and is bounding for full-power operation and partial-power operation at the corresponding MTC value allowed by the plant T/S.

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: Claudia M. Craig.

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

Date of amendment request: November 30, 1999; as supplemented on August 15, 2000.

Description of amendment request: The licensee proposed to amend the unit's Technical Specifications (TSs), Section 3.4.4, ''Emergency Ventilation System [EVS],'' and Section 3.4.5, ''Control Room Air Treatment [CRAT] System,'' to require testing consistent with American Society for Testing and Materials (ASTM) Standard D3803–1989. Currently Section 3.3.4 specifies the American National Standards Institute (ANSI) standard N510–1980. The licensee's application for amendment is a response to the NRC's Generic Letter (GL) 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.'' The staff had previously published a notice (65 FR 9009, February 23, 2000) for the licensee's November 30, 1999, submittal. The licensee's August 15, 2000, submittal revises the original submittal by increasing the charcoal bed testing efficiency of the EVS from 95 percent to 99.5 percent, and requiring the pressure drop across the CRAT System high efficiency particulate air (HEPA) filters and charcoal adsorber banks to be demonstrated to be less than 1.5 inches of water.

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

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

The proposed TS change will require the demonstration that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 1.5 inches of water at system design flow rate (± 10%). The CRAT System does not involve initiators or precursors to an accident previously evaluated, as this system performs mitigative functions in response to an accident. Failure of this system would result in the inability to perform its mitigative function, but would not increase the probability of an accident. Therefore, the probability of an accident previously evaluated is not increased.

The NMP1 [Nine Mile Point Unit 1] CRAT System is designed to limit doses to control room operators to less than the values allowed by General Design Criterion 19. This system contains HEPA filters and activated charcoal adsorber banks that are required by TS to have a combined pressure drop across them of less than 6 inches of water. The proposed TS change to require a combined pressure drop of less than 1.5 inches of water will assure the capability of the CRAT System to maintain the required minimum positive pressure in the Control Room complex. Therefore, the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed TS change will revise the allowable pressure drop across the CRAT System HEPA filters and charcoal adsorber banks to less than 1.5 inches of water at system design flow rate (± 10%). This change will not involve placing the system in a new configuration or operating the system in a different manner that could result in a new or different kind of accident. Maintaining a combined pressure drop across the HEPA filters and charcoal adsorber banks to less than 1.5 inches of water will assure system capability of maintaining the required minimum positive pressure in the Control Room complex. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed TS change will not adversely affect the performance characteristics of the CRAT System, nor will it affect the ability of the system to perform its intended function. The combined pressure drop across the CRAT System HEPA filters and charcoal adsorber banks is demonstrated to determine whether sufficient flow exists to maintain the minimum positive pressure in the control room assumed in the design basis analysis. The proposed TS change will require the combined pressure drop across the HEPA filters and charcoal adsorber banks to be less than 1.5 inches of water. This will assure system capability to maintain the required minimum positive pressure in the Control Room complex. 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: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Marsha K. Gamberoni.

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

Date of amendment request: August 16, 2000.

Description of amendment request: The amendment would (1) remove ''Offgas Treatment System Explosive Gas Mixture Instrumentation,'' Specification 3.7, from the Radiological Effluent Technical Specifications (RETS) contained in Appendix B and include reference to the Offgas Treatment System Explosive Gas Monitoring Program in Administrative Section 6 to the Technical Specifications contained in Appendix A; (2) replace the position title of Radiological and Environmental Services Manager, contained in the Administrative Section 6 of Appendix A, with radiation protection manager; and (3) revise Plant Staff organization requirements contained in Administrative Section 6 to require either the Operations Manager or Assistant Operations Manager hold a senior reactor operator license.

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:

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

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

The proposed changes simplify the RETS and meet Code of Federal Regulation requirements as specified in 10 CFR 50.36. Future changes to these requirements will be controlled by 10 CFR 50.59. The proposed changes are administrative in nature and do not involve any modification to any plant equipment or effect plant operation. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of any previously evaluated accident.

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

The proposed changes are administrative in nature, do not involve any physical alterations to any plant equipment, and cause no change in the method by which any safety related system performs its function. Therefore, this proposed amendment will not create the possibility of a new or differen[t] kind of accident from any accident previously evaluated.

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

The proposed changes are administrative in nature, will not alter the basic regulatory requirements, and do not affect any safety analyses. Therefore, no margin of safety is reduced as a result of these changes.

Based on the above evaluation, the Authority has concluded that these changes do not involve a significant hazards consideration.

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

Attorney for licensee: Mr. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Marsha K. Gamberoni.

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

Date of amendment request: August 28, 2000.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 5.5.2 b. ''Primary Coolant Sources Outside Containment'' by changing the system leak test frequency from a ''refueling cycle'' to ''at least once every 18 months.'' The proposed change will also allow the provisions of Surveillance Requirement (SR) 3.0.2 to apply to TS 5.5.2 b. (SR 3.0.2 allows a surveillance to be preformed within 1.25 times the interval specified in a Frequency.)

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

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

No. The proposed changes affect programmatic administrative controls of the Vogtle Electric Generating Plant (VEGP) Technical Specifications (TS) for leak testing systems or portions thereof that are outside containment and could contain highly radioactive fluids. Only the interval for leak testing is affected by the proposed change, and this interval has no impact on the likelihood of any of the initiating events assumed for any accident previously evaluated. Therefore, the proposed change will not result in a significant increase in the probability of any accident previously evaluated. Whereas the current TS require testing at refueling cycle intervals or less, the proposed change will specify testing at least once per 18 months, and the provisions of Surveillance Requirement (SR) 3.0.2 will be applicable. Refueling cycle intervals at VEGP are nominally 18 months in duration, but they can vary with unplanned outages, power reductions, etc. Under the proposed change, leak testing will be performed at 18-month intervals, regardless of actual refueling cycle length, and if an extension of that interval becomes necessary for systems or portions thereof due to scheduling considerations, the provisions of SR 3.0.2 will provide the necessary flexibility. However, the maximum extension that can be applied is 25% of 18 months or four and one-half months. Leak testing will continue at regular intervals, and any necessary maintenance to minimize leakage will continue to be performed. Therefore, the proposed change will not result in a significant increase in the consequences of any accident previously evaluated.

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

No. The proposed change affects only the interval at which leak test requirements are performed pursuant to TS 5.5.2.b. The proposed change does not alter the operation of the plant or any of its equipment, introduce any new equipment, or result in any new failure mechanisms or limiting single failures. Therefore, there is no potential for a new accident and no changes to the way that an analyzed accident will progress. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The proposed change affects only the interval at which leak test requirements are performed pursuant to TS 5.5.2.b. Under the proposed change, leak testing will be performed at 18-month intervals, regardless of actual refueling cycle length, and if an extension of that interval becomes necessary for systems or portions thereof due to scheduling considerations, the provisions of SR 3.0.2 will provide the necessary flexibility. However, the maximum extension that can be applied is 25% of 18 months or four and one-half months. Leak testing will continue at regular intervals, and any necessary maintenance to minimize leakage will continue to be performed. The intent of the program is maintained while providing the same scheduling flexibility that is already provided for the surveillance requirements of Section 3.0 of the TS. Therefore, the proposed change will not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard L. Emch, Jr.

Tennessee Valley Authority, Docket Nos. 50–390 and 50–391, Watts Bar Nuclear Plant Units 1 and 2, Rhea County, Tennessee

Date of amendment request: March 10, 2000.

Description of amendment request: The proposed amendment would change the Operating License to Physical Security/Contingency Plan—Tamper Indicating/Line Supervision Alarms Testing Frequency at Watts Bar Nuclear Plant (WBN) Units 1 and 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.

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

There are no safety-related systems, components, or radiological waste systems associated with the tamper indicating/line supervision alarms. The proposed change to the Physical Security Plan does not involve any physical alterations of plant configuration, changes to setpoints, or changes to any operating parameters of the security system. The proposed change does not increase the frequency of the precursors to design basis events or operational transients analyzed in the Watts Bar Final Safety Analysis Report. * * * Consequently, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no safety-related systems, components, or radiological waste systems associated with the tamper indicating/line supervision alarms. The proposed change to extend the testing frequency cannot create a Final Safety Analysis Report type accident. * * * Consequently, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Implementation of this activity will not reduce the margin of safety in the Technical Specification as there are no Technical Specification requirements associated with the physical security system. The proposed amendment to the Physical Security Plan does not change or reduce the effectiveness of any security/safeguards measures currently in place at WBN. * * * 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. The staff has also reviewed the changes to License Condition 2.E for Watts Bar Unit 1 Operating License, as well as the change to the Security Plan. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

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

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

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

Date of amendment request: May 12, 2000.

Brief description of amendment request: The proposed amendment would revise the standard to which the control room ventilation charcoal and supplementary leak collection and release system (SLCRS) charcoal must be laboratory tested as specified in: BVPS–1 Technical Specification (TS) 4.7.7.1.1.c.2 for the control room emergency habitability systems; BVPS–1 TS 4.7.8.1.b.3 for the SLCRS; BVPS–2 TS 4.7.7.1.d for the control room emergency air cleanup and pressurization system; and BVPS–2 TS 4.7.8.1.b.3 for the SLCRS. NRC Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal,'' dated June 3, 1999, requested licensees to revise their TS criteria associated with laboratory testing of ventilation charcoal to a valid test protocol, which included American Society for Testing and Materials (ASTM) D3803–1989. This license amendment request revises the charcoal laboratory standard to follow ASTM D3803–1989 for each BVPS Unit. This license amendment request also: (1) Revises the minimum amount of output in kilowatts needed for the control room emergency ventilation system heaters at each BVPS unit; (2) revises BVPS–1 SLCRS surveillance testing criteria to be consistent with American Nuclear Standards Institute/American Society of Mechanical Engineers N510–1980, the BVPS–1 control room ventilation testing, and BVPS–2 SLCRS/control room ventilation testing; and (3) makes minor typographical corrections and editorial changes.

*Date of publication of individual notice in **Federal Register:*** August 29, 2000 (65 FR 52449).

Expiration date of individual notice: September 28, 2000.

FirstEnergy Nuclear Operating Company, et al., Docket No.50–412, Beaver Valley Power Station, Unit No. 2, Shippingport, Pennsylvania

Date of amendment request: May 1, 2000, as supplemented July 21, 2000.

Brief description of amendment request: The proposed amendment would: (1) Revise Technical Specification (TS) requirements regarding the minimum number of radiation monitoring instrumentation channels required to be operable during movement of fuel within the containment; (2) revise the Modes in which the surveillance specified by Table 4.3–3, ''Radiation Monitoring Instrumentation Surveillance Requirements,'' Item 2.c.ii is required; (3) revise TS 3.9.4, ''Containment Building Penetrations,'' to allow both personnel air lock (PAL) doors and other containment penetrations to be open during movement of fuel assemblies within containment, provided certain conditions are met; (4) revise applicability and action statement requirements of TS 3.9.4. to be for only during movement of fuel assemblies within containment; (5) revise periodicity and applicability of Surveillance Requirement (SR) 4.9.4.1; (6) revise SR 4.9.4.2 to verify flow rate of air to the supplemental leak collection and release system (SLCRS) rather than verifying the flow rate through the system; (7) add two new SRs, 4.9.4.3 and 4.9.4.4, for verification and demonstration of SLCRS operability; (8) modify TS 3/4.9.9 for the containment purge exhaust and isolation system to be applicable only during movement of fuel assemblies within containment; (9) revise associated TS Bases as well as make editorial and format changes; and, (10) revise the BVPS–2 Updated Final Safety Analysis Report (UFSAR) description of a fuel-handling accident (FHA) and its radiological consequences. The changes to the BVPS–2 UFSAR reflect a revised FHA analysis that the licensee performed to evaluate the potential consequences of having containment penetrations and/or the PAL open during movement of fuel assemblies within containment. These UFSAR revisions include potential exclusion area boundary, low population zone, and control room operator doses as a result of an FHA.

Date of publication of individual notice in ***Federal Register:*** August 23, 2000 (65 FR 51342).

Expiration date of individual notice: September 22, 2000.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: January 20, 2000, as supplemented April 3 and July 7, 2000.

Brief description of amendments: The amendments revised the technical specifications to extend the allowable completion times associated with restoration of an inoperable emergency diesel generator. The amendments also permitted the performance of the 24-hour endurance run during Modes 1 and 2.

Date of issuance: September 1, 2000.

Effective date: Effective upon completion of the plant modifications cited in the April 3, 2000, submittal.

Amendment Nos.: 114 and 108.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 19, 2000 (65 FR 21035). The April 3 and July 7, 2000, submittals provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 1, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 18, 2000, as supplemented by letter dated July 27, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.7.10, ''Control Room Area Ventilation System,'' and TS 3.7.12, ''Auxiliary Building Filtered Ventilation Exhaust System,'' to establish actions to be taken for inoperable ventilation systems due to a degraded control room pressure boundary or emergency core cooling system pump rooms pressure boundary, respectively.

Date of issuance: September 5, 2000.

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

Amendment Nos.: 187/180.

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

Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34744).

The supplement dated July 27, 2000, provided additional clarifications that did not change the scope of the April 18, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 2000.

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 16, 1998; supplemented January 25, August 5, and October 4, 1999; and March 29 and June 8, 2000.

Brief description of amendments: The amendments revised the Technical Specifications associated with the High Pressure Injection System.

Date of Issuance: September 6, 2000.

Effective date: As of the date of issuance and shall be implemented within 75 days.

Amendment Nos.: 314, 314, & 314.

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

Date of initial notice in ***Federal Register:*** February 24, 1999 (64 FR 9187).

The supplements dated August 5 and October 4, 1999; and March 29 and June 8, 2000, provided clarifying information that did not change the scope of the December 16, 1998, or the January 25, 1998, submittals and the proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 27, 2000.

Brief description of amendment: The amendment revised the Technical Specifications by providing actions associated with inoperable control room emergency ventilation or cooling systems during movement of irradiated fuel during shutdown modes of operation, when the allowed outage times associated with these systems are not met.

Date of issuance: August 28, 2000.

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

Amendment No.: 219.

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

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15379).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 2000.

No significant hazards consideration comments received: No .

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

Date of application for amendment: March 8, 2000, as supplemented by letters dated June 13 and August 15, 2000.

Brief description of amendment: The amendment revised Technical Specification Definition 1.12 , ''Core Alteration,'' to explicitly define core alteration as the movement or manipulation of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel.

Date of issuance: September 7, 2000.

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

Amendment No.: 220.

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

Date of initial notice in ***Federal Register:*** April 5, 2000 (65 FR 17914).

The August 15, 2000, supplement withdrew the exclusion clause, ''excluding coupling/uncoupling of control element assemblies,'' from the proposed definition in the initial application. The June 13 and August 15, 2000, supplemental letters provided clarifying information that was within the scope of the original FEDERAL REGISTER notice and did not change the staff's initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 15, 1999, as supplemented by letter dated March 29, 2000.

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

Date of issuance: September 5, 2000.

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

Amendment No.: 167.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 26, 2000, (65 FR 4275). The March 29, 2000, supplement provided clarifying information that did not expand the scope of the original **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 29, 1999, and as supplemented by letter dated June 29, 2000.

Brief description of amendment: Entergy Operations, Inc. (licensee) has proposed to revise its Updated Final Safety Analysis Report (UFSAR) to discuss the probability threshold for when physical protection of safety-related components from tornado missiles is required for certain components. The proposed changes involve the use of Nuclear Regulatory Commission (NRC) approved probability risk methodology to assess the need for additional tornado missile protection and demonstrate that the probability of damage due to tornado missiles striking safety related components is acceptably low.

Date of issuance: September 7, 2000.

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

Amendment No.: 168.

Facility Operating License No. NPF–38: The amendment revised the UFSAR.

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37426).

The June 29, 2000, supplement provided clarifying information that did not expand the scope of the original **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No

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

Date of application for amendments: May 1, 2000.

Brief description of amendments: The amendments revised the Unit 1 and 2 Technical Specification (TS) 3/4.6.4.2 Surveillance Requirement (SR). The change allows performance of hydrogen recombiner functional test at containment pressures greater than 13 psia. This is accomplished by measuring the flow under normal or current test conditions (*e.g.*, atmospheric pressure) and calculating the expected system performance under design basis operating conditions. The surveillance was revised to verify that the recombiner flow, when corrected to the post-accident design conditions, is greater than or equal to the required flow. The corresponding design basis temperature for post-accident recombiner operation is included in the SR because it is required to correct the test flow to the design basis operating conditions. In order to support the calculations necessary to confirm the recombiner blower performance, the change included the addition of an equation and associated discussion to the bases. The equation will correct the measured test flow to a corresponding flow at the design basis operating pressure and temperature. In addition to the technical change described above, SR 4.6.4.2.b.3 was modified by separating the criteria for the system blower performance and heater operation into separate parts of the same surveillance to improve the presentation of the requirements. Format and editorial changes were included as necessary to facilitate the revision of the TS text to conform to the current TS page format, and addition of text to the bases.

Date of issuance: September 7, 2000.

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

Amendment Nos.: 232 and 114.

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

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37427).

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

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC through September 22, 2000 or at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852 effective September 26, 2000, and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room).

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: January 20, 2000, as supplemented April 3 and July 7, 2000.

Brief description of amendments: The amendments revised the technical specifications to extend the allowable completion times associated with restoration of an inoperable emergency diesel generator. The amendments also permitted the performance of the 24-hour endurance run during Modes 1 and 2.

Date of issuance: September 1, 2000.

Effective date: Effective upon completion of the plant modifications cited in the April 3, 2000, submittal.

Amendment Nos.: 114 and 108.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 19, 2000 (65 FR 21035). The April 3 and July 7, 2000, submittals provided additional information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 1, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 18, 2000, as supplemented by letter dated July 27, 2000.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 3.7.10, ''Control Room Area Ventilation System,'' and TS 3.7.12, ''Auxiliary Building Filtered Ventilation Exhaust System,'' to establish actions to be taken for inoperable ventilation systems due to a degraded control room pressure boundary or emergency core cooling system pump rooms pressure boundary, respectively.

Date of issuance: September 5, 2000.

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

Amendment Nos.: 187/180.

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

Date of initial notice in ***Federal Register:*** May 31, 2000 (65 FR 34744).

The supplement dated July 27, 2000, provided additional clarifications that did not change the scope of the April 18, 2000, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 2000.

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 16, 1998; supplemented January 25, August 5, and October 4, 1999; and March 29 and June 8, 2000.

Brief description of amendments: The amendments revised the Technical Specifications associated with the High Pressure Injection System.

Date of Issuance: September 6, 2000.

Effective date: As of the date of issuance and shall be implemented within 75 days.

Amendment Nos.: 314, 314, and 314.

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

Date of initial notice in ***Federal Register:*** February 24, 1999 (64 FR 9187).

The supplements dated August 5 and October 4, 1999; and March 29 and June 8, 2000, provided clarifying information that did not change the scope of the December 16, 1998, or the January 25, 1998, submittals and the proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 27, 2000.

Brief description of amendment: The amendment revised the Technical Specifications by providing actions associated with inoperable control room emergency ventilation or cooling systems during movement of irradiated fuel during shutdown modes of operation, when the allowed outage times associated with these systems are not met.

Date of issuance: August 28, 2000.

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

Amendment No.: 219.

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

Date of initial notice in ***Federal Register:*** March 22, 2000 (65 FR 15379).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 8, 2000, as supplemented by letters dated June 13 and August 15, 2000.

Brief description of amendment: The amendment revised Technical Specification Definition 1.12 , ''Core Alteration,'' to explicitly define core alteration as the movement or manipulation of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel.

Date of issuance: September 7, 2000.

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

Amendment No.: 220.

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

Date of initial notice in ***Federal Register*** April 5, 2000 (65 FR 17914).

The August 15, 2000, supplement withdrew the exclusion clause, ''excluding coupling/uncoupling of control element assemblies,'' from the proposed definition in the initial application. The June 13 and August 15, 2000, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: July 15, 1999, as supplemented by letter dated March 29, 2000.

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

Date of issuance: September 5, 2000.

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

Amendment No.: 167.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 26, 2000 (65 FR 4275). The March 29, 2000, supplement provided clarifying information that did not expand the scope of the original Federal **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 29, 1999, and as supplemented by letter dated June 29, 2000.

Brief description of amendment: Entergy Operations, Inc. (licensee) has proposed to revise its Updated Final Safety Analysis Report (UFSAR) to discuss the probability threshold for when physical protection of safety-related components from tornado missiles is required for certain components. The proposed changes involve the use of Nuclear Regulatory Commission (NRC) approved probability risk methodology to assess the need for additional tornado missile protection and demonstrate that the probability of damage due to tornado missiles striking safety related components is acceptably low.

Date of issuance: September 7, 2000.

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

Amendment No.: 168.

Facility Operating License No. NPF–38: The amendment revised the UFSAR.

Date of initial notice in ***Federal Register:*** June 14, 2000 (65 FR 37426).

The June 29, 2000, supplement provided clarifying information that did not expand the scope of the original **Federal Register** notice, or change the scope of the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of application for amendment: June 17, 1999, as supplemented September 15, 1999, and February 15, and June 29, 2000.

Brief description of amendment: The amendment revised Technical Specifications (TSs) Section 3.4.9.1 and associated figures to extend the applicability of the heatup and cooldown curves from 10 Effective Full Power Years (EFPY) to 15 EFPY. The changes included new heatup and cooldown curves developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2, and Code Case N–640. The applicability of TS Section 3.4.9.3, Overpressure Protection Systems, was also updated to 15 EFPY, and the maximum allowable power-operated relief valve setpoints for the over pressure protection system were revised. Revisions to the TS Bases were also made.

Date of issuance: September 6, 2000.

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

Amendment No: 113.

Facility Operating License No. NPF–73. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 17, 1999, (64 FR 62707). The September 15, 1999, and February 15, and June 29, 2000, letters provided supplemental and revised information, but did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 1, 2000.

Brief description of amendments: The amendments revised the Unit 1 and 2 Technical Specification (TS) 3/4.6.4.2 Surveillance Requirement (SR). The change allows performance of hydrogen recombiner functional test at containment pressures greater than 13 psia. This is accomplished by measuring the flow under normal or current test conditions (*e.g.*, atmospheric pressure) and calculating the expected system performance under design basis operating conditions. The surveillance was revised to verify that the recombiner flow, when corrected to the post-accident design conditions, is greater than or equal to the required flow. The corresponding design basis temperature for post-accident recombiner operation is included in the SR because it is required to correct the test flow to the design basis operating conditions. In order to support the calculations necessary to confirm the recombiner blower performance, the change included the addition of an equation and associated discussion to the bases. The equation will correct the measured test flow to a corresponding flow at the design basis operating pressure and temperature. In addition to the technical change described above, SR 4.6.4.2.b.3 was modified by separating the criteria for the system blower performance and heater operation into separate parts of the same surveillance to improve the presentation of the requirements. Format and editorial changes were included as necessary to facilitate the revision of the TS text to conform to the current TS page format, and addition of text to the bases.

Date of issuance: September 7, 2000.

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

Amendment Nos.: 232 and 114.

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

*Date of initial notice in **Federal Register:*** June 14, 2000 (65 FR 37427).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 20, 1999.

Brief description of amendments: The amendments relocated the following Technical Specification (TS) items to the Licensing Requirements Manual:

In-core Detectors (Unit 1 and 2),
Chlorine Detection System (Unit 1 and 2)
Turbine Over-speed Protection (Unit 2 only),
Crane Travel Spent Fuel Pool Building (Unit 1 and 2).

Additionally, certain information on the Remote Shutdown Panel Monitoring Instrumentation was moved to the Updated Final Safety Analysis Report. Finally, additions to the TS Bases, and certain editorial and format changes were made.

Date of issuance: September 7, 2000.

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

Amendment Nos.: 233 and 115.

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

*Date of initial notice in **Federal Register:*** November 17, 1999 (64 FR 62709).

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 17, 1999, as supplemented by letters dated January 17, March 1, March 20, May 9, and August 21, 2000.

Brief description of amendment: This amendment revised multiple surveillance requirements to support a 24-month operating cycle.

Date of issuance: August 29, 2000.

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

Amendment No.: 115.

Facility Operating License No. NPF–58: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 25, 1999 (64 FR 46438).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 14, 2000.

Brief description of amendment: The amendment changes the implementation dates of the Improved Technical Specifications, previously issued by Amendment No. 91, and requirements for the Oscillation Power Range Monitor, previously issued by Amendment No. 92, from August 31, 2000, to December 31, 2000.

Date of issuance: August 29, 2000.

Effective date: As of the date of issuance to be implemented no later than December 31, 2000.

Amendment No.: 94.

Facility Operating License No. NPF–69: Amendment revised the Operating License.

Date of initial notice in ***Federal Register:*** July 27, 2000 (65 FR 46183).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 1, 1999

Brief description of amendment: The amendment revised the technical specifications (TS) to reflect relocation of fire protection requirements from the TS to the Defueled Safety Analysis Report, quality assurance audit requirements from the TS to the Quality Assurance Plan and modification of the administrative controls section of the TS to reflect the current facility organization.

Date of issuance: August 31, 2000.

Effective date: August 31, 2000, and shall be implemented no later than 60 days from the date of issuance. Implementation shall include the relocation of technical specification requirements to the appropriate licensee-controlled document as identified in the licensee's application dated December 1, 1999, and reviewed in the staff's safety evaluation dated August 31, 2000.

Amendment No.: 33.

Facility Operating License No. DPR–7: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 12, 2000 (65 FR 1927).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 1, 1999, as supplemented August 11 and September 1, 1999.

Brief description of amendments: The amendments revise the licenses to reflect changes related to the transfer of the license for the Peach Bottom Atomic Power Station, Units 2 and 3, to the extent held by Public Service Electric and Gas Company, to PSEG Nuclear Limited Liability Company.

Date of issuance: August 21, 2000.

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

Amendments Nos.: 234 and 238.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the License.

*Date of initial notice in **Federal Register:*** August 5, 1999 (64 FR 42728).

The August 11 and September 1, 1999, supplements provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 16, 2000.

No significant hazards consideration comments received: No.

PECO Energy Company, PSEG Nuclear LLC, Delmarva Power and Light Company, and Atlantic City Electric Company; Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: May 31, 2000, as supplemented August 18, 2000.

Brief description of amendments: The amendments revise the Peach Bottom Atomic Power Station, Units 2 and 3, Technical Specifications Surveillance Requirement 3.6.1.3.11 to allow a representative sample of reactor instrumentation line excess flow check valves (EFCVs) to be tested every 24 months, instead of testing each EFCV every 24 months.

Date of issuance: September 8, 2000.

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

Amendments Nos.: 235 & 239.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 9, 2000 (65 FR 48756). The August 18, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 8, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 6, 2000.

Brief description of amendment: This amendment eliminates the response time testing of the Reactor Trip System and the Engineered Safety Feature Actuation System.

Date of issuance: August 29, 2000.

Effective date: August 29, 2000.

Amendment No.: 146.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 3, 2000 (65 FR 25768).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 6, 2000.

Brief description of amendment: This amendment proposes to modify the pressure testing requirements for the American Society of Mechanical Engineers (ASME) Code portions of the diesel fuel oil system that currently require a hydrostatic test every 10 years at 110% of system design pressure. The revision would allow ASME Code Class 3 portions of the diesel fuel oil system to be pressure tested in accordance with Section XI of the Code as required by Technical Specification 4.0.5. This will permit the use of Code Case N–498–1 as accepted by Regulatory Guide 1.147, Revision 12, for assessment of the diesel fuel oil system pressure boundary integrity.

Date of issuance: August 29, 2000.

Effective date: August 29, 2000.

Amendment No.: 147.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 3, 2000 (65 FR 25768).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 5, 2000, as supplemented August 25, 2000.

Brief description of amendment: This amendment changes Technical Specification (TS) 3/4 6.1.6, including its Bases, and adds TS 6.8.4.h. The changes support the new requirements of 10 CFR 50.55a, which require licensees to update their Containment Vessel Structural Integrity Programs to incorporate the provisions of ASME Section XI, Subsection IWL (1992 Edition with 1992 Addenda) and the five additional provisions found in 10 CFR 50.55a(b)(2)(viii).

Date of issuance: September 6, 2000.

Effective date: September 6, 2000.

Amendment No.: 148.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 23, 2000 (65 FR 9010). The August 25, 2000, supplement revised the proposed wording of Bases Section 3/4.6.1.6 and TS 6.8.4.h to clarify the reporting requirements; clarification did not impact the initial no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 8, 1999 (PCN–454), as supplemented March 16 and May 24, 2000.

Brief description of amendments: The amendments revise Surveillance Requirement (SR) 3.8.1.18 of Technical Specification (TS) 3.8.1, ''A.C. Sources—Operating.'' The amendments revise the SR to read: Verify the timing of each sequenced load block is within its timer setting plus or minus 10% or plus or minus 2.5 seconds, whichever is greater, with the exception of the 5 second load group which is minus 0.5, plus 2.5 seconds, for each programmed time interval load sequence.

Date of issuance: September 1, 2000.

Effective date: September 1, 2000, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2–169; Unit 3–160.

Facility Operating License Nos. NPF–10 and NPF–15: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 1, 1999 (64 FR 67339).

The supplemental letters dated March 16 and May 24, 2000, provided clarifying information that was within the scope of the original application and **Federal Register** notice and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a

Safety Evaluation dated September 1, 2000.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: June 1, 2000.

Brief description of amendments: The amendments revise the reactor vessel pressure and temperature limit curves that are in the Technical Specifications.

Date of issuance: August 29, 2000.

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

Amendment Nos.: 222 and 163.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 28, 2000 (65 FR 39960).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 29, 2000.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: June 22, 2000.

Brief description of amendments: These amendments revise the Technical Specifications (TS) to remove the applicability of core alteration requirements from those TS that are designed to mitigate the consequences of a fuel handling accident. The applicable TS bases are also revised.

Date of issuance: August 28, 2000.

Effective date: August 28, 2000.

Amendment Nos.: 260 and 251.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the TS.

*Date of initial notice in **Federal Register:*** July 26, 2000 (65 FR 46017).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 28, 2000.

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: April 10, 2000, as supplemented August 9, 2000.

Brief description of amendment: Revision of Technical. Specifications (TS) to allow use of the F-star (F*) alternate repair criterion for degraded steam generator tubes.

Date of issuance: September 8, 2000.

Effective date: September 8, 2000.

Amendment No.: 27.

Facility Operating License No. NPF–90: Amendment revises the TS.

*Date of initial notice in **Federal Register:*** May 31, 2000 (65 FR 34750).

The August 9, 2000, letter provided clarifying information that 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 September 8, 2000.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 25, 2000.

Brief description of amendments: The amendments revise the Technical Specifications (TS) to allow certain reactor containment building penetrations to be open during refueling activities under appropriate administrative controls. Specifically, this revision fully adopts the NRC-approved TS Task Force (TSTF) Traveler TSTF–312, Revision 1, by adding a Note to TS 3.9.4.c denoting this provision, to clarify the use of this allowance.

Date of issuance: September 5, 2000.

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

Amendment Nos.: 78 and 78.

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

*Date of initial notice in **Federal Register:*** July 12, 2000 (65 FR 43053).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 2000.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 14, 1999.

Brief description of amendment: The amendment relocates procedural details related to the Radiological Environmental Technical Specifications (TSs) to certain licensee-controlled documents.

Date of Issuance: August 24, 2000.

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

Amendment No.: 193.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 9, 2000 (65 FR 6412).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated August 24, 2000.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 14th day of September 2000.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 00–24021 Filed 9–19–00; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Staff Meetings Open to the Public: Final Policy Statement

AGENCY: Nuclear Regulatory Commission.

ACTION: Final Policy Statement.

SUMMARY: The Nuclear Regulatory Commission is finalizing revisions to its ''Policy Statement on Staff Meetings Open to the Public,'' to state that public notice of meetings will be provided primarily through the NRC Web site at *http://www.nrc.gov.* NRC will also discontinue announcing public meetings, changes, and cancellations through its public meeting notice system electronic bulletin board, and telephone recording, and through the Weekly Compilation of Press Releases and posting in the NRC's Public Document Room.

EFFECTIVE DATE: September 20, 2000.

FOR FURTHER INFORMATION CONTACT: Rosetta O. Virgilio, Office of the Executive Director for Operations, U.S.