shipments, and the likely consequences and severity of the accidents. About 1300 shipments of spent nuclear fuel have been made in NRC-certified packages, with an exceptional safety record of no releases from accidents. Despite the previous studies and safety record, some stakeholders may have questions or concerns regarding spent nuclear fuel transport package safety. Several groups have criticized NRC's cask standards and the modal study as being insufficient to adequately demonstrate safety during severe transportation accidents.

The objective of the public meetings is to bring together representatives of the interests affected by the study to discuss their views on the issues in a ''roundtable'' format. In order to have a manageable discussion, the number of participants around the table will, of necessity, be limited. The Commission, through the facilitator for the meeting, will attempt to ensure participation by the broad spectrum of interests at the meetings, including citizen and environmental groups, nuclear industry interests, state, tribal, and local governments, experts from academia, or other agencies. Other members of the public are welcome to attend, and the public will have the opportunity to comment on each of the agenda items slated for discussion by the roundtable participants. Questions about participation may be directed to the facilitator, Francis X. Cameron.

The meetings will have a pre-defined scope and agenda focused on the major technical issues in regard to spent nuclear fuel cask performance during transportation accidents. However, the meeting format will be sufficiently flexible to allow for the introduction of additional related issues that the participants may wish to raise. The purpose of the meetings is to hear the views of the participants on the issues and options to resolve the issues for the forthcoming study. The agenda for the meetings is set forth below.

Agenda

Introductions and Welcome
  E. William Brach, Director, Spent Fuel Project Office, NRC
  Susan F. Shankman, Deputy Director, Spent Fuel Project Office, NRC
Ground Rules, Agenda Overview, Introduction of Participants
  Francis X. Cameron, Facilitator
Overview of NRC Studies on Transportation Risk
  NRC Staff
NRC Plans for the Modal Study Update
  Robert Lewis, NRC
General Overview of the Study Updates
  Sandia National Laboratories
Discussion of Issues
  Participants and Audience
Summary and Closing Remarks

Dated at Rockville, Maryland, this 14th day of October, 1999.

For the Nuclear Regulatory Commission.

Susan F. Shankman,

Deputy Director, Licensing and Inspection Directorate, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 99–27362 Filed 10–19–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of October 18, 25, November 1, and 8, 1999.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of October 18

Wednesday, October 20

9:25 a.m. Affirmation Session (Public meeting) (if needed)
9:30 a.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public meeting) (Contact: Paul Lohaus, 301–415–3340)

Thursday, October 21

9:30 a.m. Briefing on Part 35—Rule on Medical Use of Byproduct Material (Public Meeting) (Contact: Cathy Haney, 301–415–6825) (SECY–99–201, *Draft Final Rule—10 CFR Part 35, Medical Use of Byproduct Material,* is available in the NRC Public Document Room or on NRC web site at: ''www.nrc.gov/NRC/COMMISSION/SECYS/index.html'' Download the *zipped version* to obtain all attachments.)

Week of October 25—Tentative

There are no meetings scheduled for the Week of October 25.

Week of November 1—Tentative

Thursday, November 4

9:25 a.m. Affirmation Session (Public Meeting) (if needed)
9:30 a.m. Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360)

Week of November 8—Tentative

Tuesday, November 9

9:00 a.m. Meeting on NRC Interactions with Stakeholders on Nuclear Materials and Waste Activities (Public Meeting)

Wednesday, November 10

9:25 a.m. Affirmation Session (Public Meeting) (if needed)
9:30 a.m. Briefing on Draft Maintenance Regulatory Guide (Public Meeting) (Contact: Richard Correia, 301–415–1009)

* The schedule for commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: Bill Hill (301) 415–1661.

* * * * *

The NRC Commission Meeting Schedule can be found on the Internet at:

http://www.nrc.gov/SECY/smj/schedule.htm

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This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to it, please contact the Office of the Secretary, Attn: Operations Branch, Washington, DC 20555 (301–415–1661). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to wmh@nrc.gov or dkw@nrc.gov.

Dated: October 15, 1999.

William M. Hill, Jr.,

SECY Tracking Officer, Office of the Secretary.

[FR Doc. 99–27459 Filed 10–18–99; 10:46 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section

189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 25, 1999, through October 7, 1999. The last biweekly notice was published on October 6, 1999 (64 FR 54370).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 19, 1999, 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 and at the local public document room for the particular facility involved. 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: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, 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, and at the local public document room for the particular facility involved.

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

Date of amendments request: September 14, 1999

Description of amendments request: Request No. 1: The proposed administrative change to Technical Specification (TS) 5.5.2, Primary Coolant Sources Outside Containment, would delete the references to the post-accident sampling return piping of the radioactive waste gas system and the post-accident sampling return piping of the liquid radwaste system because the Palo Verde post-accident sampling system does not have return lines to the radioactive waste gas or liquid radwaste systems.

Request No. 2: This proposed TS amendment would also delete the administrative requirement in TS 5.6.2, Annual Radiological Environmental Operating Report, that states: ''[t]he report shall identify the TLD [thermoluminescence dosimeter] results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.''

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:

Request No. 1

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

No—This proposed administrative change to Technical Specification (TS) 5.5.2 to delete references to the radioactive waste gas system and liquid radwaste system in the context of the post accident sampling system (PASS) does not involve a significant increase in the probability or consequences of an accident previously evaluated. Leak testing requirements of the PASS return piping are included in the TS 5.5.2 requirements that are not being changed. The appropriate PASS piping, including return piping, is leak tested per the prescribed requirements in TS 5.5.2. This administrative change would simply clarify TS 5.5.2, since the PASS return piping is not part of the waste gas or liquid radwaste systems. There is no physical connection between the PASS piping and the radioactive waste gas or liquid radwaste systems. The radioactive waste gas system and the liquid radwaste system are not part of PASS and would not contain highly radioactive fluids during a serious transient or accident to be subject to TS 5.5.2. This administrative change would involve no change to the design or maintenance of the plant and no changes in the functional requirements of any system.

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

No—This proposed administrative change to delete references to the radioactive waste gas system and liquid radwaste system in the context of PASS does not create the possibility of a new or different kind of accident from any accident previously evaluated. Leak testing requirements of the PASS return piping are implicitly included in the TS 5.5.2 requirements that are not being changed. The appropriate PASS piping, including return piping, is leak tested per the prescribed requirements in TS 5.5.2. There is no physical connection between the PASS piping and the radioactive waste gas or liquid radwaste systems. The radioactive waste gas system and the liquid radwaste system are not part of PASS and would not contain highly radioactive fluids during a serious transient or accident to be subject to TS 5.5.2. This administrative change would involve no change to the design or maintenance of the plant and no changes in the functional requirements of any system. This administrative change would simply clarify TS 5.5.2, since the PASS return piping is not part of the waste gas or liquid radwaste systems.

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

No—This proposed administrative change does not involve a significant reduction in a margin of safety. There is no margin of safety associated with this proposed administrative change to Technical Specification 5.5.2. Leak testing requirements of the PASS return piping are implicitly included in the TS 5.5.2 requirements that are not being changed. The appropriate PASS piping, including return piping, is leak tested per the prescribed requirements in TS 5.5.2. This administrative change would involve no change to the design or maintenance of the plant and no changes in the functional requirements of any system. This administrative change would simply clarify TS 5.5.2, since the PASS return piping is not part of the waste gas or liquid radwaste systems.

Request No. 2

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

No—This proposed administrative change to Technical Specification (TS) 5.6.2 does not involve a significant increase in the probability or consequences of an accident previously evaluated. This proposed TS amendment would delete the administrative requirement in TS 5.6.2, Annual Radiological Environmental Operating Report, that states: ''[t]he report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.'' The NRC ended their TLD program at the end of 1997. The requirements of TS 5.6.2 and the changes being made with this request are purely administrative reporting requirements that have no effect on the design, operation, or maintenance of the plant. Since there is no effect on the design, operation, or maintenance of the plant, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No—This proposed administrative change to TS 5.6.2 does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change only affects administrative reporting requirement and has no effect on the design, operation, or maintenance of the plant. Since this proposed change is purely administrative and would have no effect on the design, operation, or maintenance of the plant, this change will not create possibility of a new or different type of accident than any previously evaluated.

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

No—This proposed administrative change to TS 5.6.2 does not involve a significant reduction in a margin of safety. This TS establishes requirements for reporting radiological monitoring information to the NRC. Since TS 5.6.2 contains an administrative reporting requirement, and this proposed change would simply delete an administrative requirement associated with a discontinued NRC monitoring program, there is no margin of safety associated [with] this TS or with the proposed changes to the requirements of TS 5.6.2. Also, since this involves only administrative reporting, this change has no [e]ffect on any other margin of safety.

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

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999

NRC Section Chief: Stephen Dembek

CBS Corporation (Licensee), Westinghouse Test Reactor, Waltz Mill Site, Westmoreland, Pennsylvania, Docket No. 50–22, License No. TR–2

Date of amendment request: September 7, 1999, as supplemented on October 1, 1999

Description of amendment request: CBS Corporation is the licensee for the Westinghouse Test Reactor (WTR) at Waltz Mill, Pennsylvania. The licensee is authorized to only possess the reactor and a decommissioning plan has been approved. The licensee is planning to revise the decommissioning plan by reassigning the responsibilities of the Site Manager, who works for the Westinghouse Electric Company (a contractor to CBS) to the TR–2 Decommissioning Project Director who works for CBS.

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 proposed amendment to a license of a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in the margin of safety.

The staff agrees with the licensee's no significant hazards consideration determination submitted on September 7, 1999, for the following reason:

In order to complete the decommissioning of the WTR facility as described in the Decommissioning Plan, CBS has established contractual agreements with the Westinghouse Electric Company to supply continued site support and services to the Westinghouse Test Reactor Facility. CBS has also entered into contracts with other third party organizations as described in the Decommissioning Plan. These contracts will remain in place between CBS and each respective third party so that there will be no effective change in the personnel associated with the on-going decommissioning project under the TR–2 License. CBS continues to retain full responsibility for the project.

The only change being made is that the responsibilities of the Westinghouse Electric Company Site Manager, as it pertains to the WTR and the TR–2 License, has been assigned to the TR–2 Decommissioning Project Director, who works for CBS. The Westinghouse Electric Company personnel who reported to the Site Manager will now report directly to CBS through the contract.

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

The proposed amendment does not modify the WTR facility configuration or licensed activities. Thus no new accident initiators are introduced. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in the margin of safety.

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

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

Date of amendment request: September 16, 1999.

Description of amendment request: The amendments would revise Surveillance Requirements (SRs) 3.8.4.8 and 3.8.4.9 of the Technical Specifications and Bases SR 3.8.4.8 to allow testing of the direct current (DC) channel batteries with the units on line. The proposed change to SR 3.8.4.8 would also prohibit the diesel generator (DG) batteries from being service tested while the units are on line.

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:

First Standard

Implementation of this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. Approval of this amendment will have no significant effect on accident probabilities or consequences. The 125 Volt DC Vital Instrumentation and Control Power System is not an accident initiating system; therefore, there will be no impact on any accident probabilities by the approval of this amendment. The design of the system is not being modified by this proposed amendment. It has been shown that the required battery testing can be performed safely with the unit on line well within the allowed outage time for an inoperable DC channel. Both safety trains would continue to be capable of performing their required design functions in the event of an accident. Therefore, there will be no impact on any accident consequences.

Second Standard

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

Third Standard

Implementation of this amendment would not involve a significant reduction in a margin of safety. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these fission product barriers will not be impacted by implementation of this proposed amendment. It has already been shown that both safety trains of the 125 Volt DC Vital Instrumentation and Control Power System will continue to be able to perform their accident mitigation functions should they be required. In addition, the probabilistic risk analysis conducted for this proposed amendment demonstrated that there is no appreciable increase in overall plant risk incurred by its implementation. No safety margins will be impacted.

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

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina.

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

NRC Section Chief: Richard L. Emch, Jr.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of amendment request: July 29, 1999, as supplemented by letter dated August 30, 1999.

Description of amendment request: The proposed amendment would delete a license condition that required installation of a neutron flux monitoring system, in the form of excore wide range monitors (WRM), in conformance with Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' WNP–2 installed the WRM system in the spring of 1989. Removal of the license condition would allow WNP–2 to deactivate the WRM system. *Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. As stated in the NRC safety evaluation approving NEDO–31558–A (Reference 2) [in licensee's August 30,1999 letter], Category 1 neutron flux monitoring instrumentation is not needed for existing BWRs to cope with Loss-of-Coolant Accident (LOCA), Anticipated Transient Without SCRAM (ATWS), or other accidents that do not result in severe core damage conditions. Instrumentation to monitor the progression of core melt accidents would best be addressed by the current severe accident management program. Also, WRM is not included in the WNP–2 IPE/PSA models and WRM is not relied upon for operator actions in the Emergency Operating Procedures (EOPs) or actions accounted for in Severe Accident Management. Therefore, no individual precursors of an accident are affected and the elimination of the WRM does not impact or change the probabilities of accidents previously evaluated. In addition, since the operability of plant systems designed to mitigate accident consequence has not changed, the consequences of an accident previously evaluated are not expected to increase.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in procedures that may create the potential for new or different personnel errors. The elimination of the WRM system does not create the possibility of a new or different kind of accident because plant crews are trained to use the Neutron Monitoring System (NMS) in normal evolutions and under emergency conditions according to EOP guidance. In addition, NEDO–31558–A concludes that the failure of all neutron flux monitoring instrumentation does not prevent the operator from determining the shutdown condition of the reactor. Sufficient information is available on which to base operational decisions and to conclude that reactivity control has been accomplished. For example, Rod Position Information System (RPIS) is powered from an uninterruptible source and remains available even during Station Blackout (SBO) conditions to provide full core control rod position information as a backup reactor power indicator based on calculations of rod worth and shutdown margin. The proposed change does not introduce any new modes of operation or alter system setpoints which could create a new or different kind of accident. Therefore, no new precursors of an accident and no new or different kinds of accidents are created.

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

The elimination of the WRM system does not result in a reduction of the margin of safety. The neutron power indications necessary for operator response to ATWS are provided by the NMS not WRM. Based on a WNP–2 specific evaluation against the alternate criteria specified in NEDO–31558–A, there is sufficient confidence that the instrumentation would still be available to confirm that the reactor is shutdown. In addition, failure of the existing neutron flux monitoring instrumentation does not prevent plant operators from determining the shutdown condition of the reactor. Sufficient information is available to the operator to make operational decisions and to conclude that reactivity control has been accomplished. The proposed changes will not impact the basis for any Technical Specification related to the establishment or maintenance of nuclear safety margins. Therefore, operation of the facility in accordance with the proposed amendment does not involve a reduction in a margin of safety.

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

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Attorney for licensee: Perry D. Robinson, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, D.C. 20005–3502.

NRC Section Chief: Stephen Dembek.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: February 19, 1999.

Description of amendment request: The proposed amendment would revise the Crystal River Unit 3 Improved Technical Specifications Sections 5.6.2.7, 5.6.2.8, and 5.7.2.b, related to the Containment Tendon Surveillance Program. The proposed changes are a result of revisions to 10 CFR 50.55a which are required to be fully implemented by September 9, 2001. These revised requirements affect the surveillance methods for the containment tendons and the conduct of containment visual inspections, and the methods of reporting the results of the required inspections to the NRC.

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

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

No. The proposed change to the Crystal River Unit 3 (CR–3) Improved Technical Specifications (ITS) replaces the previous programmatic commitment to implement a Containment Tendon Surveillance Program based on Regulatory Guide 1.35, Revision 3, with a Containment Inspection Program that complies with the current requirements of 10 CFR 50.55a. Effective September 9, 1996, 10 CFR 50.55a requires licensees to implement a Containment Inspection Program in compliance with the 1992 Edition with the 1992 Addenda of Subsection IWE, ''Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants,'' and with Subsection IWL, ''Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,'' of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with additional modifications and limitations as stated in 10 CFR 50.55a(b)(2)(ix). Florida Power Corporation (FPC) is implementing a Containment Inspection Program to comply with these new regulatory requirements. The final rule specifies requirements to assure that the critical areas of the containment structure are routinely inspected to detect and take corrective action for defects that could compromise structural integrity. This proposed ITS change is requested to update the ITS to these latest 10 CFR 50.55a regulatory requirements.

By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. Maintaining containment structural integrity is independent of the operation of the reactor coolant system (RCS), and independent of the reactor protection system (RPS) and emergency core cooling system (ECCS). The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. By assuring the effectiveness of this barrier through appropriate inspection, and by implementing corrective actions for any degradation discovered during these inspections that might lead to containment structural failures, the probability or consequences of accidents will not be greater than that previously evaluated.

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

No. Maintaining containment structural integrity is independent of the operation of the RCS, and independent of the RPS and ECCS. By implementing corrective actions for any degradation discovered during the required inspections of the containment, the possibility of a new or different kind of accident will not be created.

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

No. The margin of safety as defined by the CR–3 ITS has not been reduced. By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. By implementing the Containment Inspection Program, the existing margin of safety is preserved.

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.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Attorney for licensee: R. Alexander Glenn, General Counsel (MAC–BT15A), Florida Power Corporation, P. O. Box 14042, St. Petersburg, Florida 33733–4042.

NRC Section Chief: Sheri R. Peterson.

GPU Nuclear, Inc. et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of amendment request: July 7, 1999.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to change the component surveillance frequencies for the following TSs to indicate a frequency of once per 3 months: Core Spray System TS 4.4.A.1 and 4.4.A.2, Containment Cooling System TS 4.4.C.1, Emergency Service Water System TS 4.4.D.1, Fire Protection System TS 4.4.F (isolation valves only), and Pressure Suppression Chamber—Drywell Vacuum Breakers TS 4.5.F.5.a. The TSs currently stipulate a component surveillance frequency of once per month. Also, the amendment would revise TS pages 4.4–1 and 4.4–2 to incorporate editorial format changes and TS page 4.4–3 to accommodate the expanded text.

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

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

The proposed surveillance interval change does not alter the actual surveillance requirements, nor does it alter the limits and restrictions on plant operations. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not degraded by the proposed change to the surveillance interval. Assurance of system and equipment availability is maintained. The proposed change does not alter any system or equipment configuration.

Based on the above, the proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed surveillance interval change does not alter the actual surveillance requirements, nor does it alter the limits and restrictions on plant operations. Assurance of system and equipment availability is maintained. The proposed change does not alter any system or equipment configuration nor does it introduce any new mechanisms which could contribute to the creation of a new or different kind of accident than previously evaluated.

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

The proposed change extends the surveillance interval for verifying the operability of the specified pumps and valves from once per month to once per three months. The proposed change does not alter the actual surveillance requirements, the limits and restriction on plant operations nor the design, function or manner of operation of any structures, systems or components. System availability and reliability are maintained. Accordingly, the proposed TS 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.

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: S. Singh Bajwa.

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 17, 1999.

Description of amendment requests: The proposed amendments would allow credit in the applicable subcriticality analysis for the negative reactivity provided by insertion of the rod cluster control assemblies (RCCAs) during realignment from a cold leg recirculation to a hot leg recirculation configuration. This realignment, which is referred to as hot leg switchover, is performed following a loss-of-coolant accident. This methodology change, when evaluated in accordance with 10 CFR 59.59, resulted in an unreviewed safety question that will require prior approval by the NRC staff in accordance with the provisions of 10 CFR 50.90

prior to implementation. The proposed change would also affect the Bases for Technical Specification (T/S) 3/4.5.5, ''Refueling Water Storage Tank,'' and several sections of the Updated Final Safety Analysis Report (UFSAR).

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?

No. I&M [Indiana Michigan Power Company] proposes to credit RCCA insertion of negative reactivity for criticality control during the core cooling flow path realignment from cold leg recirculation to hot leg recirculation following the postulated cold leg LBLOCA [large-break loss-of-coolant accident]. No physical modifications will be made to plant systems, structures, or components.

Credit for RCCAs is only being applied to demonstrate core subcriticality upon hot leg switchover (HLSO) following a cold leg LBLOCA. The performance criteria codified in 10 CFR 50.46 continue to be met. The ability of the RCCAs to insert under LOCA and seismic conditions was a function important to safety as part of the original CNP [Cook Nuclear Plant] design basis. This is supported by the conclusion presented in NRC (at the time, the Atomic Energy Commission) Safety Evaluation Report (SER), Section 3.3, ''Mechanical Design of Reactor Internals,'' dated January 14, 1969. The SER includes the statements that, ''[t]he control rod guide tubes are designed so that each finger of each control rod assembly is always partially inserted in the guide tube. Deflection limits on the guide tubes have been chosen so that deflections caused by blow-down forces during a loss-of-coolant accident will not prevent control rod insertion,'' and that the ''* * * mechanical design of internals, fuel assemblies, and control elements is acceptable.'' However, the licensing basis safety analyses for the LBLOCA scenario have conservatively not taken credit for insertion of the RCCAs.

No physical modifications will be made to plant systems, structures, or components in order to implement the proposed methodology change. The safety functions of the safety related systems and components, which are related to accident mitigation, have not been altered. Therefore, the reliability of RCCA insertion is not affected. As such, taking credit for RCCA insertion does not alter the probability of an LBLOCA (the design basis accident at issue). The Westinghouse analyses provided as Attachments 6 and 7 [to the licensee's application] demonstrate that RCCA insertion will occur, with substantial margin, following a design basis cold leg LBLOCA combined with a seismic event. Crediting RCCA insertion does not affect mechanisms for a malfunction that could impact the HLSO subcriticality analysis, or mechanisms that could initiate a LOCA. Taking credit for the negative reactivity available from insertion of the RCCAs, which is currently assumed for various accident analyses within the CNP licensing basis (e.g., small break LOCA, main steamline break, feedline break, steam generator tube rupture), does not affect equipment malfunction probability directly or indirectly. Therefore, crediting the RCCAs as a source of negative reactivity for post-LOCA criticality control at the time of HLSO does not significantly increase the probability of an accident previously evaluated.

Furthermore, the traditional conservative assumption that the most reactive RCCA is stuck fully out of the core is being maintained. A malfunction that results in one RCCA to fail to insert is a credible scenario, and is being considered for the post-LOCA subcriticality analysis following a cold leg LBLOCA. There will be sufficient negative reactivity, even with the most reactive RCCA stuck fully out of the core, to assure core subcriticality post-LOCA, as supported by the subcriticality analysis that is confirmed each and every fuel cycle as part of the reload documentation (i.e., the Reload Safety Evaluations). The core is shown to remain subcritical during the post-LOCA long-term cooling period, specifically while HLSO is performed. Thus, no additional radiological source terms are generated, and the consequences of an accident previously evaluated in the UFSAR will not be significantly increased.

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

No. The proposed change involves crediting the negative reactivity that is available from the RCCAs for an analysis applicable several hours after the initiation of a cold leg LBLOCA. As such, this change involves post-LOCA recovery actions several hours after the break has occurred and does not involve accident initiation. As discussed above, the original design requirements for the CNP reactor internals, core fuel assemblies, and RCCAs were based upon assuring the ability of the RCCAs to insert following a double-ended rupture LOCA with seismic loadings. Thus, the safety functions of safety related systems and components have not been altered by this change. Crediting the negative reactivity that is available from the RCCAs for the post-LOCA subcriticality analysis upon HLSO does not cause the initiation of any accident, nor does the proposed activity create any new credible limiting single failure. Crediting the insertion of RCCAs does not result in any event previously deemed incredible being made credible nor is there any introduction of any new failure mechanisms that are not currently considered in the design basis LOCA. There are no changes introduced by this amendment concerning how safety related equipment is designed to operate under normal or design basis accident conditions since the calculations supporting RCCA insertion following a cold leg LBLOCA have assumed design basis break sizes in conjunction with seismic loadings. Therefore, the possibility of an accident of a different type than already evaluated in the UFSAR is not created.

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

No. Presently, no credit is taken for RCCA insertion in the analysis to demonstrate post-cold leg LOCA subcriticality at the time of HLSO. The current subcriticality analysis for this scenario relies only on the boron provided by the RWST [refueling water storage tank] and the accumulators. Thus, RCCA insertion provides another source of negative reactivity (margin of safety). Revising the post-cold leg LBLOCA HLSO subcriticality analysis to credit the negative reactivity associated with the RCCAs is a means to offset the sump dilution associated with the effects of the inactive regions of the CNP containment sump. The incorporation of this ''defense-in-depth'' source of negative reactivity in the HLSO subcriticality analysis has been conservatively determined to cause a reduction in the margin of safety. 10 CFR 50, Appendix K, I.A.2., states, in part, that ''[r]od trip and insertion may be assumed if they are calculated to occur,'' and provides for crediting RCCA insertion as an acceptable feature of emergency core cooling system (ECCS) evaluation models. The proposed change is based upon an analysis for CNP that demonstrates that the control rods will indeed insert and the resulting negative reactivity can be credited for post-LOCA criticality control.

The proposed change would ensure that post-LOCA subcriticality is maintained during HLSO. Subsequently, there would not be a challenge to long-term core cooling due to a return to a critical condition. This being the case, the requirements of 10 CFR 50.46(b)(5) that, ''* * * the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time* * *'' continues to be satisfied and the margin of safety in the CNP licensing basis is preserved. 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.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

Attorney for licensee: Jeremy J. Euto, Esq., 500 Circle Drive, Buchanan, MI 49107.

NRC Section Chief: Claudia M. Craig.

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: September 29, 1999.

Description of amendment request: The proposed amendment requests a Technical Specification change that would extend the allowed out-of-service time for the residual heat removal service water system (RHRSW) from 7 days to 11 days on a one-time basis while modifications are made on the RHRSW ''A'' strainer.

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:

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

The Conditional Core Damage Probability due to this proposed change is calculated to be 6.4 E–8. This value falls below the threshold probability of 1 E–6 for risk significance of temporary changes to the plant configuration in the EPRI PSA [Electric Power Research Institute Probability Assessment] Applications Guide (Reference 3) [see application dated September 29, 1999].

This proposed change does not increase the consequences of an accident previously evaluated because all relevant accidents (LOCA) [loss-of-coolant accident] would result in the transfer of decay heat to the suppression pool. For this scenario, the same complement of equipment will be available to achieve and maintain cold shutdown as is required by the current Technical Specification LCO [limiting condition for operation].

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

The proposed change does not physically alter the plant. As such, no new or different types of equipment will be installed. The new design for the RHRSW strainer packing gland will be evaluated under a separate 10 CFR 50.59 evaluation and is considered to be functionally equivalent for the purposes of this one-time-only proposed Technical Specification change.

The implementation and use of the contingency plan for achieving limited containment heat removal in the event the B division of RHRSW is rendered inoperable will be evaluated under the Authority's 10 CFR 50.59 program.

Involve a significant reduction in a margin of safety.

The Conditional Core Damage Probability due to this proposed change is calculated to be 6.4 E–8. This value falls below the threshold probability of 1 E–6 for risk significance of temporary changes to the plant configuration in the EPRI PSA Applications Guide (Reference 3).

The consequences of a postulated accident occurring during the extended allowable out-service time are bounded by existing analyses therefore there is no 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.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

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

NRC Section Chief: S. Singh Bajwa.

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

Date of amendment request: December 1, 1998, as supplemented by letters of April 21, 1999, and July 19, 1999.

Description of amendment request: The proposed amendments would revise the Technical Specifications to reflect replacing the current Model 51 steam generators with Westinghouse Model 54F steam generators. The replacement program includes re-analyzing and evaluating loss-of-coolant-accident (LOCA) and non-LOCA mass and energy releases, containment and sub-compartment pressure and temperature responses, dose analyses, and the effects on nuclear steam supply and balance of plant systems.

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

1. The proposed changes do not significantly increase the probability or consequences of an accident previously evaluated in the [Final Safety Analysis Report] FSAR. The comprehensive engineering effort performed to support [steam generator] SG replacement has included evaluations or re-analysis of all accident analyses including all dose related events. All dose consequences have been analyzed or evaluated with respect to these proposed changes, and all acceptance criteria continue to be met. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident than any accident already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed technical specification changes have no adverse effects on any safety-related system and do not challenge the performance or integrity of any safety-related system. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed technical specification changes do not involve a significant reduction in a margin of safety. All applicable analyses supporting the [steam generator] SG replacement reflect these proposed values. All acceptance criteria (including LOCA peak clad temperature, [departure from nucleate boiling] DNB, containment temperature and pressure, and dose limits) continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed Southern Nuclear Company'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.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of application for amendments: June 30, 1999 (TS 98–10).

Description of amendment requests: The proposed amendments would change the Sequoyah (SQN) Operating Licenses DPR–77 (Unit 1) and DPR–79(Unit 2) by updating the current Technical Specification requirements for reactor coolant system leakage detection and operational leakage specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), 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.

The proposed revisions enhance the Technical specification (TS) requirements to provide greater consistency with the standard TS in NUREG–1431. This revision proposes changes to the requirements for reactor coolant system (RCS) leak detection and RCS operational leakage in Specifications 3.4.6.1 and 3.4.6.2, respectively. New Specifications

3.4.6.3 and 3.5.6 for RCS pressure isolation valves and emergency core cooling system (ECCS) seal injection flow have been added to improve consistency with NUREG–1431. The proposed revisions are not the result of changes to plant equipment, system design, testing methods, or operating practices. The modified requirements will allow some relaxation of current operability criteria, action requirements, and surveillance requirements (SRs). These changes provide more appropriate requirements in consideration of the safety significance and the design capabilities of the plant as determined by the improved standard TS industry effort. These specifications serve to primarily provide identification and control of the RCS fission product barrier leakage and ECCS degradation and are not considered to be a contributor to the generation of postulated accidents. Since these proposed revisions will continue to support the required safety functions, without modification of the plant features, the probability of an accident is not increased.

The proposed changes will allow relaxation of action times for inoperable leak detection features and the components that can be inoperable. The required actions to ensure acceptable pressure isolation valve capability with an inoperable valve have been revised to allow isolation by a single valve for a limited period of time. These revisions will allow unit operation for a longer period of time with reduced system redundancy. However, the redundancy reduction and action time increases are not significant and will continue to provide an acceptable level of safety considering the significance of RCS leakage, other design features or compensatory actions that provide equivalent functions, and the unlikely chance of an event that would require functions for leakage identification during the proposed time interval. These considerations are consistent with the basis developed by the industry and NRC for NUREG–1431. Surveillances have been removed from the RCS operational leakage specification as a result of relocated requirements, duplication of other SRs, and testing requirements that do not provide a significant benefit in the identification of RCS leakage. The SRs that have been retained or relocated to other TS specifications will provide acceptable verifications for the timely identification of conditions that indicate an unacceptable amount of RCS leakage or potential ECCS degradation resulting from excessive seal injection flow.

The limiting condition for operation associated with the seal injection flow requirements has been revised to utilize a modified operability criteria. The proposed change will provide a range of differential pressures and the corresponding seal flows that would be representative of the existing single point flow limit. This change does not alter the intent of the operability requirements, but does allow the flexibility to use equivalent values that provide the same level of assurance for ECCS operability. The proposed operability condition for seal injection flow enhances the current requirement by establishing additional test parameters that will ensure that the amount of seal injection flow does not degrade the ECCS functions.

The proposed changes to the SQN TS provide flexibility without modifying the functions of required safety systems. In many instances the proposed changes ensure that plant conditions for surveillance testing are more appropriate for testing purposes and the verification of system operability.

These changes are consistent with the intent of NUREG–1431 and result in the enhancement of the SQN TSs based on the latest industry and NRC positions. The provisions proposed in this change request will continue to maintain an acceptable level of protection for the health and safety of the public and will not significantly impact the potential for the offsite release of radioactive products. The overall effect of the proposed change will result in specifications that have equivalent or improved requirements compared to existing specifications for RCS leakage and ECCS operability and will not significantly increase the consequences of an accident.

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

The proposed revisions are not the result of changes to plant equipment, system design, testing methods, or operating practices. The modified requirements will allow some relaxation of current operability criteria, action requirements, and SRs consistent with NUREG–1431. These changes provide more appropriate requirements in consideration of the safety significance and the design capabilities of the plant as determined by the improved standard TS industry effort. These specifications serve to primarily provide identification and control of the RCS fission product barrier leakage and ECCS degradation and are not considered to be a contributor to the generation of postulated accidents. Since the functions of the associated systems will continue to perform without change and were not previously considered to contribute to accident generation, the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed changes, associated with RCS leakage and ECCS functions, will not result in changes to system design or setpoints that are intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems. These systems will continue to operate without change and only the associated actions or testing activities have been altered. Revisions to the actions and surveillances provide some relaxation and flexibility such that longer intervals are allowed for inoperable components and testing requirements are revised to provide conditions that provide more accurate results. The increased action times are acceptable considering the available redundant features, the compensatory measures provided by the actions, and the allowed time intervals that have been developed by the industry and NRC and recommended in NUREG–1431. The SR changes actually provide test condition requirements that enhance the accuracy of the activity even though they may allow a delay in the performance of the test. These surveillance changes are also in accordance with NUREG–1431 recommendations.

These revisions will continue to provide the necessary actions to minimize the impact of inoperable equipment to an acceptable level and will provide testing activities that will ensure system operability. Since the setpoints and design features that support the margin of safety are unchanged and actions for inoperable systems continue to provide appropriate time limits and compensatory measures, the proposed changes will not significantly reduce the margin of safety.

The NRC 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.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

NRC Section Chief: Sheri R. Peterson.

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

Date of amendment request: September 28, 1999 (TS 99–007).

Description of amendment request: The proposed amendment on Response Time Test (RTT) elimination would revise the Watts Bar Nuclear Plant Unit 1 Technical Specifications (TS) definitions for ''Engineered Safety Feature (ESF) Response Time'' and ''Reactor Trip System (RTS) Response Time'' to provide for verification of response time for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC. In addition, associated changes to the Bases for Surveillance Requirements would also be made.

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.

This change to the TS does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS and ESF instrumentation is being used, the time response allocations/modeling assumptions in the Chapter 15 analyses are unchanged; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR [Updated Final Safety Analysis Report]. Therefore, the proposed amendment 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.

This change does not alter the performance of pressure and differential pressure transmitters, process protection racks (Eagle 21), nuclear instrumentation (NIS), and logic system (SSPS) used in the plant protection systems. These components/systems will still have response time verified by test prior to placing the equipment in operational service and after any maintenance that could affect the response time of that equipment. Changing the method of periodically verifying instrument response time for applicable instrumentation from RTT to calibration and channel checks or functional test will not create any new accident initiators or scenarios. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and pressure differential sensors, Eagle 21, NIS, and SSPS is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response time is within that assumed in the safety analysis, since calibration checks and functional tests will detect any degradation which might significantly affect equipment response time. Therefore, the proposed license amendment request does not result in a significant reduction in margin of safety.

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

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, TN 37402.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Sheri Peterson.

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

Date of amendment request: June 15, 1999.

Description of amendment request: The licensee proposed revisions to Technical Specifications (TSs) Sections 3.1/4.1 Reactor Protection System and 3.2/4.2 Protective Instrument Systems instrumentation, tables, and the associated bases to increase the surveillance test intervals (STIs), add allowable out-of-service times (AOTs), replace generic ECCS actions for inoperable instrument channels with function-specific actions, and relocate selected trip functions from the TSs to a Vermont Yankee (VY) controlled document. In addition, revision to TS Section 3.1/4.1 Reactor Protection System and the associated bases is proposed to remove the RUN Mode APRM Downscale/IRM High Flux/Inoperative Scram Trip Function (APRM Downscale RUN Mode SCRAM). The submittal also proposes to implement editorial corrections and administrative changes that do not alter the meaning or intent of the requirements.

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

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

VY has determined that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The generic analysis contained in Licensing Topical Report NEDC–30851P–A assessed the impact of changing SCRAM (RPS) surveillance test intervals for Logic and Functional tests (STIs) and adding allowable out-of-service times (AOTs) on the SCRAM (RPS) failure frequency, the scram frequency and equipment cycling. Specifically, Section 5.7.4, ''Significant Hazards Assessment,'' of NEDC–30851P–A states that:

''Fewer challenges to the safeguards system, due to less frequent testing of the RPS, conservatively results in a decrease of approximately one percent in core damage frequency. This decrease is based upon the following:

Based on the plant-specific experience presented in Appendix J, the estimated reduction in scram frequency (0.3 scrams/yr.) represents a 1 to 2 percent decrease in core damage frequency based on the BWR plant-specific Probabilistic Risk Assessments (PRAS) listed in Table 5–8.

The increase in core damage frequency due to less frequent testing is less than one percent. This increase is even lower (less than 0.01 percent) when the changes resulting from the implementation of the Anticipated Transients Without Scram (ATWS) rule are considered. Therefore, this increase is more than offset by the decrease in CDF due to fewer scrams.

The effect of reducing unnecessary cycles on RPS equipment, although not easily quantifiable, also results in a decrease in core damage frequency.

The overall impact on core damage frequency of the changes in allowable out-of-service times is negligible.''

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated, namely the increase in probability of a scram failure due to SCRAM (RPS) unavailability is insignificant, and the overall probability of an accident is actually decreased as the time the SCRAM (RPS) Instrumentation logic operates as designed is increased resulting in less inadvertent scrams during testing and repair. Furthermore, the plant specific reports demonstrate[ ] that although VY differs from the generic model analyzed in License Topical Report NEDC–30851P–A, the net effect of the plant-specific differences do not alter the generic conclusions.

The generic analysis contained in Licensing Topical Reports NEDC–30851P–A Suppl 2/NEDC–31677P–A assessed the impact of changing STIs and AOTs for BWR Isolation Instrumentation common/not common to SCRAM (RPS) and ECCS instrumentation. Specifically, Section 4.0, ''Summary of Results,'' of NEDC–30851P–A Suppl 2 states that:

''The results indicate that the effects on probability of failure to initiate isolation are very small and the effects on probability or frequency of failure to isolate are negligible in nearly every case. In addition, the results indicate that increasing the AOT to 24 hours for tests and repairs has a negligible effect on the probability of failure of the isolation function. These combined with changes to the testing intervals and allowed out-of-service times for RPS and ECCS instrumentation provide a net improvement to plant safety and operations.''

and Section 5.6, ''Assessment of Net Effect of Changes,'' of NEDC–31677P–A states that:

''A reduction in core damage frequency (CDF) of at least as much as estimated in the ECCS instrumentation analysis can be expected when the isolation actuation instrumentation STIs are changed from one month to three months. The chief contributor to this reduction is the channel functional tests for the MSIVs. Inadvertent closure of the MSIVs will cause an unnecessary plant scram. This reduction in CDF more than compensates for any small incremental increase (10% or 1OE–07/year) in calculated isolation function failure frequency when the STI is extended to three months.''

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the consequences of an accident previously evaluated, namely the increase in probability of an isolation failure due to isolation instrumentation unavailability is insignificant, and the overall probability of an accident is actually decreased as the time the SCRAM (RPS) Instrumentation logic operates as designed is increased resulting in less inadvertent scrams during testing and repair.

The generic analysis contained in Licensing Topical Report NEDC–30936P–A (Parts 1 and 2) assessed the impact of changing STIs and AOTs for all BWR ECCS Actuation Instrumentation. Specifically, Section 4.0, ''Technical Assessment of Changes,'' of NEDC–30936P–A (Part 2) states that:

''The results indicate an insignificant (less than 5E–7 per year) increase in water injection function failure frequency when STIs are increased from 31 days to 92 days, AOTs for repair of the ECCS actuation instrumentation are increased from one hour to 24 hours, and AOTs for surveillance testing are increased from two to six hours. For all four BWR models the increase represents less than 4% increase in failure frequency. However, when other factors which influence the overall plant safety are considered, the net result is judged to be an improvement in plant safety.''

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated, namely the increase in probability of a water injection failure due to ECCS instrumentation unavailability is insignificant and the net result is judged to be an improvement in plant safety. Furthermore, the plant specific report demonstrates that although VY differs from the generic model analyzed in Licensing Topical Report NEDC30936P–A, the net affect of the plant-specific differences do not alter the generic conclusions.

The generic analysis contained in Licensing Topical Report NEDC–30851 P–A Supp 1, assessed the impact of changing Rod Block STIs on Rod Block failure frequency. Specifically, Section 5 (BNL's Tech. Eval. Report—Attach. 2 to the NRC SER) of NEDC–30851 P–A Suppl 1 states that:

''The BWR Owners'' Group proposed changes to the Technical Specifications concerning the test requirements for BWR control rod block instrumentation. The changes consist of increasing the surveillance test intervals from one to three months. These test interval extensions are consistent with the already approved changes to STIs for the reactor protection system. The technical analysis reviewed and verified as documented herein indicates that there will be no significant changes in the availability of the control rod block function if these changes are implemented. In addition, there will be a negligible impact on the plant core melt frequency due to the decreased testing.''

From this generic analysis, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability of an accident previously evaluated or consequences of an accident previously evaluated.

Bases contained in GE Topical Report GENE–770–06–1 assessed the impact of changing STIs and AOTs on selected systems failure frequency. Specifically, Section 2.0, ''Summary,'' of GENE 770–06–1 states that:

''Technical bases are provided for selected proposed changes to the instrumentation STIs and AOTs that were identified in the BWROG Improved BWR Technical Specification activity. These STI and AOT changes are consistent with approved changes to the RPS, ECCS, and isolation actuation instrumentation. These proposed changes do not result in a degradation to overall plant safety.''

From these Bases, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability of an accident previously evaluated or consequences of an accident previously evaluated.

Bases contained in GE Topical Report GENE–770–06–2 assessed the impact of changing STIs and AOTs on selected systems (RCIC Actuation) failure frequency. Specifically, Section 2.0, ''Summary,'' of GENE 770–06–2 states that:

''The STI and AOT changes to the RCIC actuation instrumentation are justified based on their small effect on the water injection function unavailability and consistency with comparable changes to the actuation instrumentation for the other ECCS subsystems''. These STI and AOT changes are consistent with approved changes to the RPS, ECCS, and isolation actuation instrumentation. These proposed changes do not result in a degradation to overall plant safety.''

From these Bases, the BWR Owners' Group concluded that the proposed changes do not significantly increase the probability of an accident previously evaluated or consequences of an accident previously evaluated.

The proposed change will not alter the physical characteristics of any plant systems or components and all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by this change, thereby assuring that the design capabilities of those systems and components are not challenged in a manner not previously assessed so as to create the possibility of a new or different kind of accident.

The addition of allowable out-of-service times (AOTs) and the increase in surveillance test intervals (STIS) does not alter the function of the SCRAM (RPS), ECCS, Isolation, Rod Block, and Selected Instrument Systems nor involve any type of plant modification and no new modes of plant operation are involved with these changes.

No physical change is being made to any systems or components that are credited in the safety analysis, therefore there is no change in the probability or consequences of any accident analyzed in the UFSAR.

The design basis accident applicable to the startup power region is the Control Rod Drop Accident (CRDA). The UFSAR does not credit the RUN Mode IRM High Flux/ Inoperative with the associated APRM downscale scram Trip Function (APRM downscale RUN Mode SCRAM) in the termination of this accident, Accident mitigation is provided by the APRM 120% power scram. Therefore, elimination of the APRM downscale RUN Mode SCRAM function has no adverse affect on previously evaluated accidents.

The Continuous Control Rod Withdrawal Error (CWE) transient is terminated by the Rod Block Monitor (RBM) in the RUN Mode. The APRM Reduced High Flux Scram provides the primary STARTUP Mode protection in conjunction with the IRMs and limits the consequences of this transient. Therefore, elimination of the APRM downscale RUN Mode SCRAM function has no effect on the consequences of this transient.

Adding a new surveillance to verify SRM/ IRM/APRM will enhance neutron monitoring during startups and shutdowns and does not have an adverse affect on previously evaluated accidents.

None of the proposed changes will affect any of the rod blocks or other precursor events to either the CRDA or CWE. Therefore, there is no change in the probability of any accident previously analyzed.

Use of ECCS Function-specific AOTs, actions and relocation of Bus Power Monitors to a licensee controlled document is consistent with STS and does not have an adverse affect on previously evaluated accidents.

In addition, VY concluded the editorial corrections and administrative changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. These changes do not alter the meaning or intent of any requirements.

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

VY has determined that the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will not alter the physical characteristics of any plant systems or components and all safety-related systems and components remain within their applicable design limits. Thus, system and component performance is not adversely affected by this change, thereby assuring that the design capabilities of those systems and components are not challenged in a manner not previously assessed so as to create the possibility of a new or different kind of accident. Editorial corrections and administrative changes do not alter the meaning or intent of any requirements.

The addition of allowable out-of-service times (AOTs), ECCS function-specific actions and the increase in surveillance test intervals (STIs) does not alter the function of the SCRAM (RPS), ECCS, Isolation, Rod Block, and Selected Instrument Systems nor involve any type of plan modification and no new modes of plant operation are involved with these changes. Therefore, operation in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Elimination of APRM downscale RUN Mode SCRAM function affects only the operations of neutron monitoring and protective systems (IRM and APRM) which provide indication and mitigation actions only. Operation of these systems does not create the possibility for new precursors (such as reactivity) which would introduce a new or different kind of accident from any accident previously evaluated.

Additionally, the proposed changes do not affect the ability of those systems required to mitigate previously evaluated accidents during the modes they are credited.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety The NRC staff has reviewed and approved the generic studies contained in the GE Topical Reports (LTRs) and has concurred with the BWR Owners' Group that the proposed changes do not significantly affect the availability of the SCRAM (RPS), ECCS, Isolation, Rod Block, or Selected Instrument Systems. The proposed addition of allowable out-of-service times (AOTs) for the instruments addressed in the LTRs provide reasonable time for making repairs and performing tests. The lack of sufficient AOTs in the current Technical Specifications (TS) creates a hurried atmosphere during repairs and tests that could cause an increased risk of error. In addition, placing an individual channel in a tripped condition because no AOT exists, as in the current TS, increases the potential of an inadvertent scram. The proposed AOTs provide realistic times to complete the required actions without increasing the overall instrument failure frequency. Use of ECCS Function-specific AOTs, actions and relocation of Bus Power Monitors to a licensee controlled document is consistent with STS and there is no significant reduction in the margin of safety.

Editorial corrections and administrative changes do not alter the meaning or intent of any requirements. Therefore, there is no significant reduction in the margin of safety.

The incorporation of extended surveillance test intervals (STIs) does not result in significant changes in the probability of instrument failure, as demonstrated by the LTRs. In addition, the TS calibration frequency has not changed, and therefore assurance exists that the setpoints will not be affected by drift.

These changes, when coupled with the reduced probability of test-induced plant transients and equipment failures, result in an overall increase in the margin of safety.

The only scram function that the UFSAR takes credit for in the mitigation of the limiting accident (control rod drop accident) is the APRM 120% power scram which is not affected by this change. Only the APRM Downscale RUN Mode SCRAM, for which the UFSAR takes no credit in the termination of any analyzed event, is removed by this change. Removal of the APRM Downscale RUN Mode SCRAM will avoid the need to operate the plant in a ''half scram'' condition with the potential for an inadvertent plant transient. For these reasons, the change does not involve a significant reduction in a margin of safety.

The Continuous Control Rod Withdrawal Error (CWE) transient is terminated by the Rod Block Monitor (RBM) in the RUN Mode. When initiated from the STARTUP Mode, the consequences of a CWE are limited by the APRM Reduced High Flux scram in conjunction with the IRM scram function. Therefore eliminating the TS requirement for the APRM Downscale RUN Mode SCRAM will not reduce the margin of safety for this transient.

Adding a new surveillance to verify SRM/IRM/APRM overlap will enhance neutron monitoring during startups and shutdown, and consequently does not involve a significant reduction in a margin of safety.

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10 CFR 50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) 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.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 21, 1999.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.10.C, ''Diesel Fuel'' by increasing the minimum usable volume of diesel fuel in the diesel fuel oil storage tank (FOST). The specified minimum amount of diesel fuel is that quantity necessary to support diesel generator operation for a period of 7 days.

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

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

The diesel generators are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. This change does not challenge or degrade the performance of any safety system assumed to function in the accident analysis. Since this change simply increases the minimum volume of stored diesel generator fuel in the FOST, its impact is to enhance the long-term operation of diesel generators used to mitigate the consequences of accidents.

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

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

This change does not affect the design or mode of operation of any plant system, structure or component. No physical alteration of plant structures, systems or components is involved, and no new or different type of equipment will be installed. Thus, no new condition of operation is created. The change is conservative in that it results in a net increase in the minimum required diesel fuel oil stored in the FOST.

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

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

The[ ] proposed change does not adversely affect a margin of safety because increasing the minimum required volume of fuel oil provides additional assurance of diesel generator availability and, therefore, maintains or increases the availability of the onsite power supply. Since this change simply increases the quantity of diesel fuel oil available for diesel generator operation, there is no reduction in any value, condition, or range of parameters used in any accident analysis.

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.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 21, 1999.

Description of amendment request: The proposed amendment would extend the effective full implementation date by six months, from December 31, 1999, to June 30, 2000, for Amendment 120 issued March 22, 1999. Amendment 120 approved a modification to the plant to increase the storage capacity of the spent fuel pool and increase the nominal fuel enrichment to 5 weight percent U-235. The extension is due to delays fabricating and installing the new spent fuel storage racks.

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

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

The proposed change is administrative in nature and does not significantly affect any system that is a contributor to initiating events for previously evaluated accidents. The proposed change does not significantly affect any system that is used to mitigate any previously evaluated accidents. Therefore, the proposed change does not involve any significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change is administrative in nature and does not alter the design, function, or operation of any plant component and does not install any new or different equipment. Therefore, a possibility of a new or different kind of accident from those previously analyzed has not been created.

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

The proposed change is administrative in nature and does not involve a significant reduction in the margin of safety associated with the fuel cladding, reactor coolant boundary, containment, or any safety limit.

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.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037.

NRC Section Chief: Stephen Dembek.

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.

Consolidated Edison Company of New York, Docket No. 50–003, Indian Point Nuclear Generating Station, Unit No. 1, Buchanan, New York

Date of amendment request: July 20, 1999.

Description of amendment request: The amendment would revise the Technical Specifications to change the senior reactor license requirement for the Operations Manager.

*Date of publication of individual notice in **Federal Register**:* September 9, 1999 (64 FR 49027).

Expiration date of individual notice: October 12, 1999.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York.

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

Date of amendment request: September 24, 1999.

Description of amendment request: The proposed amendment would revise current Technical Specification (TS) 3.6.1.8 by adding footnote "**" to Action b. The footnote would allow continued operation of Fermi 2 with the leakage of penetration X–26 exceeding the limit in TS 4.6.1.8.2, provided certain compensatory measures are taken. Operation would be allowed to continue until the next plant shutdown.

Because the NRC staff issued the Fermi 2 improved standard TSs (ITS) on September 30, 1999, with implementation within 90 days, the licensee also provided a version of the TS amendment that would be compatible with the ITS. This version would add a new special operations TS, ITS 3.10.8, to address the compensatory actions and other requirements associated with penetration X–26.

*Date of publication of individual notice in **Federal Register**:* October 1, 1999 (64 FR 53421).

Expiration date of individual notice: Comment period expires October 15, 1999; Opportunity for hearing period expires November 1, 1999.

Local Public Document Room location: Monroe County Library System, Ellis Reference and Information Center, 3700 South Custer Road, Monroe, Michigan 48161.

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, and at the local public document rooms for the particular facilities involved.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: May 20, 1999, as supplemented by letters dated September 8, 1999, September 16, 1999, and September 20, 1999.

Brief description of amendments: The amendments revised Technical Specification (TS) Section 3.8.A, ''Containment Cooling Service Water System,'' (CCSW) to clarify that only one pump is required to support operability of the Control Room Emergency Ventilation System (CREVS).

Date of issuance: October 1, 1999.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 174 and 170.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register***: August 25, 1999 (64 FR 46426). The September 8, September 16, and September 20, 1999, submittals provided additional 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 October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of application for amendments: June 15, 1999.

Brief description of amendments: The amendments revised Technical Specification (TS) 4.7.D.6 by replacing the leakage limit of 11.5 standard cubic feet per hour (scfh) for each main steam isolation valve (MSIV) with a limit of 46 scfh on the total combined leakage for the MSIVs of all four main steam lines.

Date of issuance: October 1, 1999.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 175 and 171.

Facility Operating License Nos. DPR–19 and DPR–25: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register***: July 14, 1999 (64 FR 38024).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: May 19, 1999.

Brief description of amendments: The amendments relocated Technical Specification 3/4.4.4, ''Chemistry,'' from the TS to the Updated Final Safety Analysis Report (UFSAR) and to an Administrative Technical Requirement that has been incorporated into the UFSAR by reference.

Date of issuance: October 1, 1999.

Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: 134 and 119.

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

*Date of initial notice in **Federal Register***: July 14, 1999 (64 FR 38024).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Jacobs Memorial Library, 815 North Orlando Smith Avenue, Illinois Valley Community College, Oglesby, Illinois 61348–9692.

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: May 11, 1999, as supplemented by letter dated July 13, 1999.

Brief description of amendments: The amendments revised the Technical Specifications by incorporating changes to the pressure-temperature limits; the heatup, cooldown, and inservice test limits for the reactor coolant system to a maximum of 33 Effective Full Power Years; the low temperature overpressure protection system; and operational requirements for the reactor coolant pumps.

Date of Issuance: October 1, 1999.

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

Amendment Nos.: Unit 1–307; Unit 2–307; Unit 3–307.

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

*Date of initial notice in **Federal Register***: June 16, 1999 (64 FR 32289).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: April 7, 1999, as supplemented by letters dated May 25, June 21, August 2, and August 30, 1999.

Brief description of amendment: The amendment revises the minimum critical power ratio safety limits.

Date of issuance: September 27, 1999.

Effective date: September 27, 1999.

Amendment No.: 158.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register***: May 19, 1999 (64 FR 27329).

The May 25, June 21, August 2 and August 30, 1999, supplemental letters provided additional clarifying information that did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Energy Northwest, Docket No. 50–397, WNP–2, Benton County, Washington

Date of application for amendment: April 20, 1999, as supplemented by letter dated September 9, 1999.

Brief description of amendment: The amendment revised Technical Specification 3.4.11, ''RCS Pressure and Temperature (PT) Limits,'' for 32 effective full power years (EFPY) using the latest vessel beltline material and fluence data.

Date of issuance: October 6, 1999.

Effective date: October 6, 1999.

Amendment No.: 159.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* May 19, 1999 (64 FR 27330).

The September 9, 1999, supplemental letter provided additional clarifying information, did not significantly expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

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

Date of amendment request: May 14, 1999, as supplemented by letters dated June 17, and September 7, 15, 17, and 24, 1999.

Brief description of amendment: The amendment revises the Technical Specification requirements affecting the surveillance criteria for that portion of the once-through steam generator tubes regarded as a primary-to-secondary pressure boundary located within the upper tubesheet and impacted by a specific degradation mechanism, namely, outside diameter intergranular attack.

Date of issuance: October 4, 1999.

Effective date: As of the date of issuance and shall be implemented prior to startup from the Unit 1 Cycle 15 refueling outage.

Amendment No.: 202.

Facility Operating License No. DPR–51: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* June 2, 1999 (64 FR 29709).

The June 17, and September 7, 15, 17, and 24, 1999, letters provided clarifying and additional information that did not change the scope of the May 14, 1999, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

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

Date of amendment request: July 2, 1998, as supplemented by letters dated July 7 and August 24, 1999.

Brief description of amendment: The amendment changes the ACTION requirements for Technical Specification (TS) 3/4.3.2 for the Emergency Feedwater Actuation Signal (EFAS). This change revises the allowed outage time for a channel of EFAS to be in the tripped condition from ''prior to entry into the applicable MODE(S) following the next COLD SHUTDOWN'' to the more restrictive time limit of 48 hours and adds a shutdown requirement. Additionally, the TS 3.0.4 exemption is removed from the ACTION statement for the tripped condition. Changes to TS Bases Section 3/4.3.2 are also included to support the changes.

Date of issuance: October 6, 1999.

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

Amendment No.: 154.

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

*Date of initial notice in **Federal Register**:* December 16, 1998 (63 FR 69339). The July 7 and August 24, 1999, letters provided additional information that did not change the scope of the July 2, 1998, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Florida Power and Light Company, et al., Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: February 23, 1999.

Brief description of amendment: This amendment removes redundant boron concentration monitoring requirements specified for Modes 3 through 6 contained in TS 3/4.1.2.9, ''Reactivity Control Systems-Boron Dilution.''

Date of Issuance: October 4, 1999.

Effective Date: October 4, 1999.

Amendment No.: 104.

Facility Operating License No. NPF–16: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* August 25, 1999 (64 FR 46440).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954–9003.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Nuclear Generating Plant, Unit 3, Citrus County, Florida

Date of application for amendment: May 5, 1999, as supplemented May 21, May 28, August 20, and September 2, 1999.

Brief description of amendment: Changes the Crystal River Unit 3 Technical Specifications to allow an alternate repair criteria (ARC) for axial tube end crack-like indications in the upper and lower tubesheets of the Once-Through Steam Generators (OTSGs). The ARC will allow leaving OTSG tubes with axially oriented tube end cracks located within the clad region of the tube-to-tubesheet roll joint in service.

Date of issuance: October 1, 1999.

Effective date: October 1, 1999.

Amendment No.: 188.

Facility Operating License No. DPR–72: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* June 2, 1999 (64 FR 29710). The May 21, May 28, August 20, and September 2, 1999, supplements did not affect the original no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of application for amendment: December 29, 1998, as supplemented June 18, 1999.

Brief description of amendment: Transfer of the license for Crystal River Unit 3, to the extent it is held by the City of Tallahassee, to Florida Power Corporation.

Date of issuance: October 1, 1999.

Effective date: October 1, 1999.

Amendment No.: 189.

Facility Operating License No. DPR–31: Amendment revised the License.

Date of initial notice in ***Federal Register***: February 26, 1999 (64 FR 9544). The supplemental letter dated June 18, 1999, did not change the original proposed no significant hazards consideration determination, or expand the scope of the amendment request as originally noticed.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 8, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal River, Florida 34428.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: December 16, 1998.

Brief description of amendment: The amendment relocates Technical Specification (TS) 3/4.7.10 ''Area Temperature Monitoring,'' and the associated TS Table 3.7–3, to the Technical Requirements Manual, which is referenced in the Seabrook Station Updated Final Safety Analysis Report and is the implementing manual for the TS improvement program referenced in Section 6.7 of the TSs.

Date of issuance: October 1, 1999.

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

Amendment No.: 63.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: February 10, 1999 (64 FR 6700).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: March 27, 1998, as supplemented by letter dated June 17, 1998.

Brief description of amendment: To revise Technical Specification (TS) 3.7.6.1, Control Room Emergency Makeup Air and Filtration, and TS 3.7.6.2, Control Room Air Conditioning, to delete the restriction to suspend all operations involving positive reactivity changes during the plant conditions specified.

Date of issuance: October 5, 1999.

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

Amendment No.: 64.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: April 22, 1998 (63 FR 19973). The June 17, 1998, supplement provided clarifying information and did not change the staff's proposed no significant hazards determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

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

Date of amendment request: March 31, 1999.

Brief description of amendment: The amendment revised Sections 2.10.4, 3.1, and Table 3–3 of the technical specifications to increase the minimum required reactor coolant system (RCS) flow rate and change surveillance requirements for RCS flow rate.

Date of issuance: October 6, 1999.

Effective date: October 6, 1999, to be implemented within 30 days from the date of issuance.

Amendment No.: 193.

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

Date of initial notice in ***Federal Register***: May 19, 1999 (64 FR 27322).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102.

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: March 29, 1999, as supplemented July 21, 1999.

Brief description of amendments: The amendments delete the surveillance requirement (SR) associated only with the refuel platform fuel grapple fully retracted position interlock input, which is currently required by the Peach Bottom Atomic Power Station, Units 2 and 3, Technical Specification SR 3.9.1.1.

Date of issuance: September 24, 1999.

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

Amendments Nos.: 229 and 232.

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

Date of initial notice in ***Federal Register***: August 11, 1999 (64 FR 43774). The July 21, 1999, 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 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (Regional Depository) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: July 12, 1999, and supplemented August 30, 1999.

Brief description of amendment: The amendment changed the minimum critical power ratio safety limit and the approved methodologies referenced in the core operating limits report.

Date of issuance: October 5, 1999.

Effective date: As of date of issuance and shall be implemented prior to the start of Peach Bottom Atomic Power Station Unit No. 3, Cycle 13 operation.

Amendment No.: 233.

Facility Operating License No. DPR–56: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: August 11, 1999 (64 FR 43777). The August 30, 1999, letter provided additional 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 October 5, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (Regional Depository) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

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: September 10, 1998 (PCN–496), as supplemented July 19, 1999.

Brief description of amendments: The amendments delete Technical Specification 3.6.7 relating to hydrogen recombiners.

Date of issuance: October 7, 1999.

Effective date: October 7, 1999, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 2—159; Unit 3—150.

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

*Date of initial notice in **Federal Register**: August 11, 1999 (64 FR 43778).*

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 7, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713.

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

Date of amendments request: November 6, 1998.

Brief Description of amendments: The amendments revise the TS nuclear instrumentation system (NIS) surveillance requirements. The revised TS changes require Southern Nuclear Company to adjust the NIS power range channels only when calorimetric-calculated power is greater than the power range indicated power by more than +2 percent rated thermal power. The proposed TS changes are for both the current TS and the improved TS.

Date of issuance: October 1, 1999.

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

Amendment Nos.: 144 and 135

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

*Date of initial notice in **Federal Register***: January 27, 1999 (64 FR 4160).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama.

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 application for amendments: April 13, 1999, as supplemented by letter dated August 26, 1999.

Brief description of amendments: The amendments revise Technical Specifications (TS) to update Limiting Condition for Operation (LCO) 3.0.4 and Surveillance Requirements (SR) 3.0.4 in the existing TS to be consistent with the versions of the LCO 3.0.4 and SR 3.0.4 as they appear in Revision 1 to NUREG–1431. The proposed change also adds the words ''or that are part of a shutdown of the unit,'' to LCO 3.0.4 to allow reactor shutdowns that are not necessarily required by other TS Required Actions.

Date of issuance: September 30, 1999.

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

Amendment Nos.: Unit 1—108; Unit 2—86.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register***: August 11, 1999 (64 FR 43779). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 30, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia.

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: July 29, 1999.

Brief description of amendments: The amendments revise TS Section 3.1.7, ''Standby Liquid Control (SLC) System.'' The revision replaces ''greater than the Region B limits,'' which could be misleading, with ''within the Region B limits.''

Date of issuance: September 24, 1999.

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

Amendment Nos.: Unit 1—217; Unit 2—158.

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

*Date of initial notice in **Federal Register***: August 25, 1999 (64 FR 46449). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 24, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia.

Tennessee Valley Authority, Docket No. 50–296, Browns Ferry Nuclear Plant, Unit 3, Limestone County, Alabama

Date of application for amendment: July 28, 1999 (TS–398).

Brief description of amendment: The amendment revises the Technical Specifications (TS) to implement operability and surveillance requirements for the previously-installed Oscillation Power Range Monitor trip function.

Date of issuance: September 27, 1999.

Effective date: As of the date of issuance, to be implemented at the end of the Cycle 9 outage.

Amendment No.: 221.

Facility Operating License No. DPR–68: Amendment revises the TS.

*Date of initial notice in **Federal Register***: August 25, 1999 (64 FR 46450). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 27, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Athens Public Library, South Street, Athens, Alabama 35611.

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: February 26, 1999 (TS 98–08).

Brief description of amendments: The amendments relocate Sequoyah Nuclear

Plant Technical Specification (TS) 3.7.6, ''Flood Protection Plan,'' and its associated bases from the TS to the Technical Requirements Manual. Future changes to the Flood Protection Plan will be processed in accordance with 10 CFR 50.59.

Date of issuance: October 6, 1999.

Effective date: As of the date of issuance to be implemented no later than 45 days after issuance.

Amendment Nos.: 247 and 238.

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

Date of initial notice in ***Federal Register****:* March 24, 1999 (64 FR 14286) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 6, 1999.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1001 Broad Street, Chattanooga, Tennessee 37402.

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

Date of application for amendment: July 20, 1999, as supplemented August 13, 1999.

Brief description of amendment: The amendment modifies the operability requirements for the high pressure cooling systems—High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and Automatic Depressurization System (ADS)—and the safety and relief valves, and adds a time limitation for conducting operability testing of HPCI and RCIC.

Date of Issuance: October 1, 1999.

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

Amendment No.: 177

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

Date of initial notice in ***Federal Register****:* August 31, 1999 (64 FR 47537)

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

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

Power Station, Vernon, Vermont

Date of application for amendment: June 29, 1999 Brief description of amendment: The amendment revises the leak rate requirements for the main steam line isolation valves. Specifically, a total allowable leakage rate for the sum of the four main steam lines is established that is equal to four times the current allowable individual main steam line isolation valve leakage rate. The allowable individual main steam line isolation valve leakage rate is revised to be one half of the allowable total leakage rate.

Date of Issuance: October 1, 1999.

Effective date: 10/01/99, and shall be implemented within 30 days.

Amendment No.: 178

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

Date of initial notice in ***Federal Register****:* July 28, 1999 (64 FR 40909).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 1, 1999.

No significant hazards consideration comments received: No

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 13th day of October, 1999.

John A. Zwolinski,

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

[FR Doc. 99–27210 Filed 10–19–99; 8:45 am]

BILLING CODE 7590–01–P

POSTAL SERVICE

Notice of Meeting

AGENCY: Postal Service.

ACTION: Notice of meeting.

SUMMARY: The Postal Service will hold further meetings of a Consensus Committee to develop recommendations for revision of USPS STD 7A, which governs the design of curbside mailboxes. The committee will develop and adopt its recommendations through a consensus process. The committee will consist of persons who represent the interests affected by the proposed rule, including mailbox manufacturers, mailbox accessory manufacturers, and postal customers.

MEETING DATES: The second and third committee meetings are tentatively scheduled for November 3–4, 1999 and December 14–15, 1999.

MEETING PLACE: U.S. Postal Service Headquarters, 475 L'Enfant Plaza, SW, Washington, DC 20260.

FOR FURTHER INFORMATION CONTACT: Annamarie Gildea, (202) 268–3558.

SUPPLEMENTARY INFORMATION: Mail comments and all other communications regarding the committee to Annamarie Gildea, U.S. Postal Service Headquarters, 475 L'Enfant Plaza, SW, Room 7142, Washington, DC 20260. Committee documents will be available for public inspection and copying between 9 a.m. and 4 p.m. weekdays at the address above. Entry into U.S. Postal Service Headquarters is controlled. Persons wishing to attend the November 3–4 meeting must send a fax to Annamarie Gildea at (202) 268–5293 no later than October 29, 1999 with the person's name and organizational affiliation, if any. Persons wishing to attend the December 14–15 meeting must fax the same information to the same name and number no later than December 10, 1999. For additional information regarding the USPS STD 7A Consensus Committee, see **Federal Register** Vol 64, No. 158, p. 44681 (August 17, 1999).

Neva R. Watson,

Alternate Certifying Officer, Legislative.

[FR Doc. 99–27344 Filed 10–19–99; 8:45 am]

BILLING CODE 7710–12–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–42002; File No. SR–OPRA–99–1]

Options Price Reporting Authority; Notice of Filing of Amendment to OPRA Plan Adopting a Participation Fee Payable by Each New Party to the Plan

October 13, 1999.

Pursuant to Rule 11Aa3–2 under the Securities Exchange Act of 1934 (''Exchange Act''),[1] notice is hereby given that on August 16, 1999, the Options Price Reporting Authority (''OPRA'')[2] submitted to the Securities and Exchange Commission (''SEC'' or ''Commission'') an amendment to the Plan for Reporting of Consolidated Options Last Sale Reports and Quotation Information (''Plan''). The amendment adds provisions applicable to a participation fee payable by each

[1] 17 CFR 240.11Aa3–2.

[2] OPRA is a National Market System Plan approved by the Commission pursuant to Section 11A of the Exchange Act and Rule 11Aa3–2 thereunder. *See* Securities Exchange Act Release No. 17638 (Mar. 18, 1981).

The Plan provides for the collection and dissemination of last sale and quotation information on options that are traded on the member exchanges. The five exchanges which agreed to the OPRA Plan are the American Stock Exchange (''AMEX''); the Chicago Board Options Exchange (''CBOE''); the New York Stock Exchange (''NYSE''); the Pacific Exchange (''PCX''); and the Philadelphia Stock Exchange (''Phlx'').