For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 02–27480 Filed 10–28–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Federal Register Notice; Meeting

DATE: Weeks of October 28, November 4, 11, 18, 25, December 2, 2002.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of October 28, 2002

Wednesday, October 30, 2002

2 p.m. Discussion of security issues (Closed—Ex. 1 & 9)

Thursday, October 31, 2002

9:25 a.m. Affirmation session (Public meeting), (If needed)

9:30 a.m. Briefing on EEO program (Public meeting) (Contact: Irene Little, 301–415–7380)

2:30 p.m. Briefing on proposed rulemaking to add new section 10 CFR 50.69, ''Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors'' (Public meeting) (Contact: Eileen McKenna, 301–415–2189, or Timothy Reed, 301–415–1462)

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

Friday November 1, 2002

9 a.m. Discussion of security issues (Closed—Ex. 1)

Week of November 4, 2002—Tentative

There are no meetings scheduled for the week of November 4, 2002.

Week of November 11, 2002—Tentative

Thursday, November 14, 2002

2 p.m. Discussion of management issues (Closed—Ex. 2)

Week of November 18, 2002—Tentative

Thursday, November 21, 2002

2 p.m. Discussion of security issues (Closed—Ex. 1)

Week of November 25, 2002—Tentative

There are no meetings scheduled for the week of November 25, 2002.

Week of December 2, 2002—Tentative

Wednesday December 4, 2002

10 a.m. Briefing on decommissioning bankruptcy issues (Closed—Ex. 4 & 9)

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: R. Michelle Schroll (301) 415–1662.

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

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

Dated: October 24, 2002.

R. Michelle Schroll,

Acting Technical Coordinator, Office of the Secretary.

[FR Doc. 02–27591 Filed 10–25–02; 12:31 pm]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, October 4, 2002, through October 17, 2002. The last biweekly notice was published on October 15, 2002 (67 FR 63687).

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 Rgister a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville

Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 29, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714,[1] which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be

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[1] The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: ''In all other circumstances, such ruling body or officer shall, in ruling on—

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

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

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

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

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

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

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

granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov*.

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

Date of amendments request: September 6, 2002.

Description of amendments request: The amendments would replace the peak linear heat safety limit, in Technical Specification (TS) 2.1.1.2, ''Reactor Core SLs [Safety Limits],'' by a peak fuel centerline temperature safety limit to have a safety limit in the TSs that would not be exceeded during normal operation or anticipated operational occurrences (AOOs), in accordance with Section 50.36(c)(1)(ii)(A) of Title 10 of the Code of Federal Regulations (10 CFR).

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 does not require any physical change to plant systems, structures, or components nor does it require any change in systems or plant operations. The proposed change does not result in any change to safety analysis methods or results. The change to establish peak fuel centerline temperature as the Safety Limit is consistent with the PVNGS Units 1, 2 and 3 licensing bases for ensuring that the fuel design limits are met. Operations and analysis will continue to be in accordance with the PVNGS Units 1, 2 and 3 licensing bases. The peak fuel centerline temperature is the basis for protecting the fuel and is consistent with the safety analysis. [The peak linear heat rate and peak fuel centerline temperature safety limits are not initiators of accidents.]

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

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

The PVNGS Units 1, 2 and 3 Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses for AOOs where the peak linear heat rate may exceed the existing Safety Limit of 21 kW/ft are the control element assembly (CEA) Withdrawal events at Subcritical and Low Power conditions. The analyses for these AOOs indicate that the peak fuel centerline temperature is not exceeded. The existing safety analyses, which remain unchanged, do not affect any accident initiators that would create a new accident. [The peak linear heat rate and peak fuel centerline temperature safety limits are not initiators of accidents.]

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

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

The proposed change does not result in any change to safety analysis methods or results. Therefore, by changing the Safety Limit from peak linear heat rate to peak fuel centerline temperature the margins as established in the PVNGS Units 1, 2 and 3 Technical Specifications and UFSAR are unchanged.

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

Based on the above, APS [Arizona Public Service Company] concludes that the activities associated with the proposed amendment[s] presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: 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.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendment request: September 20, 2002.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level, for Unit Nos. 1 and 2 to add two notes to allow operational changes in the Shutdown Cooling System to support operations and testing. The changes would allow the SDC pumps to be deenergized for less than or equal to 15 minutes when switching from one train to another. The second change would allow one SDC loop to be inoperable for up to 2 hours for surveillance testing, provided that the other loop was operable and in operation.

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. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The system affected by this proposed amendment is the Shutdown Cooling (SDC) System. This system mitigates the consequences of a boron dilution event and removes decay heat from the Reactor Coolant System when the unit is in Mode 6. This proposed amendment revises the Technical Specification to allow the SDC pumps to be deenergized for less than or equal to 15 minutes to allow swapping from one operating train to another, and would allow one SDC loop to be inoperable for up to two hours for surveillance testing. Because this system is used for the mitigation of an accident, it is not an accident initiator. Therefore, the probability of an accident previously evaluated is not increased.

The only design basis accident considered in this Mode is a boron dilution event. Consideration is also given to a loss of decay heat removal in this Mode as well. Both of these conditions are evaluated in the Updated Final Safety Analysis Report (UFSAR). The evaluations consider operation of the SDC system to mitigate these conditions. Removing this system from service for a limited amount of time, with other operational restrictions, limits the consequences to those already assumed in the UFSAR. Thus, no increase in offsite dose occurs under this conditions. Therefore, the consequences of an accident previously evaluated have not increased.

Therefore, the probability or consequences of an accident previously evaluated have not significantly increased.

2. Would not create the possibility of a new or different [kind] of accident from any accident previously evaluated.

The proposed changes do not involve a significant change in the operation of the plant and no new accident initiation mechanism is created by the proposed changes. The SDC System is not being altered by this amendment request. No substantial changes are made in the way in which the SDC System is operated. The only change made would allow both SDC pumps to be deenergized to swap operating trains, and one SDC inoperable for less than two hours to allow for surveillance testing. Since the SDC System is an accident mitigating system only, changes in when this system is needed to operate cannot create a new [kind] of accident.

Therefore, the possibility of a new or different [kind] of accident from any previously evaluated is not created.

3. Would not involve a significant reduction in a margin of safety.

The margin of safety provided by the SDC System is to provide boration control and to remove decay and sensible heat from the Reactor Coolant System as described in the UFSAR. Removal of system components from service as described above, and with limitations in place to prevent boron dilution and loss of decay and sensible heat removal, does not significantly impact the margin of safety. The SDC System will continue to be able to provide its safety function under this conditions. Operators will continue to have adequate time to respond to any off-normal events. Removing the system from service, for a limited period of time, with other operational restrictions limits the consequences to those already assumed in the UFSAR. Therefore, no reduction in [a] margin of safety has occurred because the event results in the UFSAR are not changed by operation in the proposed conditions.

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

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

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: August 26, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TS) for diesel fuel oil for the plant's onsite diesel-generator power sources. The proposed changes would allow the use of an optional water and sediment content test, would relocate the specific version of certain American Society for Testing and Materials (ASTM) references to licensee controlled documents, would add several new ASTM references, and would relocate the requirement for a 10-year diesel fuel oil tank inspection and cleaning to licensee controlled documents. The licensee stated that the changes are consistent with the Standard Technical Specification Travelers (TSTF) 374, Revision 0 and TSTF 2, Revision 1. Associated changes are also proposed for the TS Bases.

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

The following discussion is a summary of the evaluation of the change contained in this proposed amendment against the 10 CFR 50.92 (c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendments would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or
3. Involve a significant reduction in a margin of safety.

First Standard

The proposed changes relocate the specific American Society for Testing and Materials (ASTM) Standard references from the Administrative Controls Section of Technical Specifications (TS) to a licensee-controlled document. Since any changes of the licensee-controlled document will be evaluated to the requirements of 10 CFR 50.59, ''Changes, tests, and experiments,'' no increase in the probability or consequences of an accident previously evaluated is involved. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to the storage tanks has expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil. The Bases for SR 3.8.3.3 (CNS) and 3.8.3.2 (MNS) are revised to indicate that the API gravity is tested in accordance with ASTM D1298 or D287.

Relocating the specific ASTM Standard references from the TS to a licensee-controlled document, allowing a water and sediment test to be performed to establish the acceptability of new fuel oil, and revising the TS Bases will not affect or degrade the ability of the emergency diesel generators (DGs) to perform their specified safety function. Fuel oil quality will continue to meet ASTM requirements.

In addition Surveillance Requirements (SR) 3.8.3.5 for McGuire and 3.8.3.6 for Catawba are revised to remove the requirement for a 10-year tank inspection and cleaning. This requirement will be moved to a licensee-controlled document. Any changes of the licensee-controlled document will be evaluated to the requirements of 10CFR 50.59 ''Changes, tests, and experiments,''.

This change will not affect or degrade the ability of the emergency diesel generators (DGs) to perform their specified safety function. Fuel oil quality will continue to meet ASTM requirements.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained.

The proposed changes do not alter or prevent the ability of structures, systems, or components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated.

Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

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

Second Standard

The proposed changes relocate the specific ASTM Standard references from the Administrative Controls Section of TS to a licensee-controlled document. In addition, the ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil. The proposed changes revise Bases B 3.8.3 to reference the current specific ASTM standards. The Bases for SRs 3.8.3.3 (CNS) and 3.8.3.2 (MNS) are revised to indicate that the API gravity is tested in accordance with ASTM D1298 or D287.

In addition Surveillance Requirements (SR) 3.8.3.5 for McGuire and 3.8.3.6 for Catawba are revised to remove the requirement for a 10-year tank inspection and cleaning. This requirement will be moved to a licensee-controlled document. Any changes of the licensee-controlled document will be evaluated to the requirements of 10CFR50.59 ''Changes, tests, and experiments,''.

The changes do not involve a physical alteration to the plant (*i.e.*, no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis or licensing basis. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

The proposed changes relocate the specific ASTM Standard references from the Administrative Control Section of TS to a licensee-controlled document. Instituting the proposed changes will continue to ensure the use of the current applicable ASTM

Standards to evaluate the quality of both new and stored fuel oil designated for use in the emergency diesels. The detail associated with the specific ASTM Standard references is not required to be in the TS to provide adequate protection of the public health and safety, since the TS still retains the requirement for compliance with the applicable ASTM standard. Changes to the licensee-controlled document are performed in accordance with the provisions of 10 CFR 50.59. Should it be determined that future changes involve a potential reduction in a margin of safety, NRC review and approval would be necessary prior to the implementation of the changes. This approach provides an effective level of regulatory control and provides for a more appropriate change control process.

The ''clear and bright'' test used to establish the acceptability of new fuel oil for use prior to the addition to storage tanks has been expanded to allow a water and sediment content test to be performed to establish the acceptability of new fuel oil. The proposed changes revise Bases B 3.8.3 to allow reference to the current ASTM standard. The Bases for SR 3.8.3.3 is revised to indicate that the API gravity is tested in accordance with ASTM D1298 or D287. The level of safety of facility operation is unaffected by the proposed changes since there is no change in the intent of the TS requirements of assuring fuel oil is of the appropriate quality for emergency DG use.

In addition Surveillance Requirements (SR) 3.8.3.5 for McGuire and 3.8.3.6 for Catawba are revised to remove the requirement for a 10-year tank inspection and cleaning. This requirement will be moved to a licensee-controlled document. Any changes of the licensee-controlled document will be evaluated to the requirements of 10CFR50.59 ''Changes, tests, and experiments''. The level of safety of the facility operation is unaffected by the proposed changes since there is no change in the intent of the SR to clean and inspect the fuel tanks.

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

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: October 15, 2001, as supplemented by letter dated August 27, 2002.

Description of amendment request: The proposed amendment request provides additional information to support a modification to Technical Specification 3.4.7 and limits Reactor Coolant System activity permitted by the ACTION statement to 60 microcuries per gram (μCi/gm) at all power levels. The letdown line break accident analysis in the Final Safety Analysis Report is also changed to reflect revised dose consequences. This notice supercedes the biweekly **Federal Register** notice dated November 28, 2001 (66 FR 59504), based on the original application dated October 15, 2001.

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

Response: The proposed change to the Technical Specifications (TS) conservatively limits Reactor Coolant System (RCS) activity permitted by Action Statement 3.4.7.a to 60 μCi/gm at all reactor power levels. The proposed change to the Final Safety Analysis Report (FSAR) Section 15.6.3.1 revises the letdown line break accident analyses.

The probability of a previously evaluated accident is not affected by this change because the pre-existing iodine spike is not an accident initiator and the new letdown line break accident analysis does not affect any plant Structure, Systems, or Component (SSC) but merely determines the consequences of the previously evaluated accident.

This TS change is conservative in that it will reduce the accident consequences for events occurring at lower power levels. The new letdown line break accident analysis meets the original Safety Evaluation Report (SER) and the current Standard Review Plan (SRP) acceptance criteria of a small fraction of the 10 CFR [Part] 100 limits.

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

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

Response: The probability of a new or different accident is not affected by this change because the new letdown line break analysis does not affect any plant Structure, Systems, or Component but merely determines the consequences of the previously evaluated accident.

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

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

Response: The TS change is more limiting in that it will reduce the accident consequences for events occurring at lower power levels.

The new letdown line break accident analysis, assuming one operating charging pump, meets the original SER and current SRP acceptance criteria of a small fraction of the 10 CFR [Part] 100 limits. This single pump analysis provides a suitable licensing basis analysis and has sufficient conservatism to accommodate two and three pump operating scenarios that may exist during the operating cycle.

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

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

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: September 27, 2002.

Description of amendment request: The proposed amendment changes Appendix B, ''Environmental Protection Plan (Non-Radiological),'' of the license by removing a parenthetical reference to a superseded section of 10 CFR 51.

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 deletes a reference to a superseded section of 10 CFR 51, ''Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,'' found in the non-radiological Environmental Protection Plans (EPPs) for Byron Station, LaSalle County Station and Quad Cities Nuclear Power Station, Units 1 and 2. The EPP (Non-Radiological) is Appendix B to the Facility Operating License. The change is administrative in nature. No physical changes to the facilities will result from the proposed change. The initial conditions and methodologies used in accident analyses remain unchanged. The proposed change does not revise or alter the design assumptions for systems or components used to mitigate the consequences of accidents. Thus, accident analyses results are not impacted by this proposed change.

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

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

The proposed change deletes a reference to a superseded section of 10 CFR 51.5. The change is administrative in nature. No physical or operational changes to the facilities will result from the proposed change.

The proposed change does not affect the design or operation of any system, structure, or component (SSC) in the plant. The safety functions of the related SSCs are not changed in any manner, nor is the reliability of any SSC reduced. The change does not affect the manner by which the facility is operated and does not change any facility, structure, system, or component. No new or different type of equipment will be installed by this proposed change.

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

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

The proposed change is administrative in nature and has no impact on the margin of safety of any Technical Specification. There is no impact on safety limits or limiting safety system settings. The change does not affect any plant safety parameters or setpoints. The proposed change deletes an inaccurate reference to a section of 10 CFR 51 that has been superseded. No physical or operational changes to the facility will result from the proposed changes.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: January 16, 2002.

Description of amendment request: The amendment would make administrative, editorial, and format (including repagination) changes to the technical specification (TS) Bases index and the Administrative Control section of TSs. Specifically, the amendments would relocate the TS Bases page listings from the TS index to a TS Bases index, and remove certain duplicative administrative requirements from Section 6, ''Administrative Controls,'' of the TSs.

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

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

No. The proposed administrative changes to the TS index and to Section 6 of the TSs do not result in changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. Also, the proposed changes do not impact the design of plant systems such that previously analyzed SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

The previously analyzed SSCs are unaffected by the proposed changes and continue to provide assurance that they are capable of performing their intended design function in mitigating the effects of design basis accidents (DBAs). As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed administrative changes do not significantly increase the probability or consequences of any accident previously evaluated.

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

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

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

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

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

No. The proposed administrative changes do not affect any previously evaluated accident. The proposed changes do not adversely affect the TS requirements and will continue to ensure that the necessary plant equipment is operable in the plant conditions where these systems are required to operate to mitigate a DBA as described in the analyses presented in the UFSAR. Thus, the proposed administrative, editorial, and format changes do not affect plant safety.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 23, 2002.

Description of amendment request: The proposed amendment would revise the Unit 2 reactor coolant system (RCS) pressure-temperature curves in Technical Specification (TS) Figures 3.4–2 and 3.4–3 and associated TS Bases. The revised curves will bound operation of the unit for the remainder of its current license duration and bound operation with planned license amendments to increase the power level at which the unit is allowed to operate.

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated

The proposed change will revise the RCS pressure-temperature curves to bound operation of the reactor for up to 32 EFPY at a power level of up to 3800 MW for the current fuel cycle and beyond, to reflect new fluence analysis methodology, to reflect the use of ASME [American Society of Mechanical Engineers] Code Case N–641, to include boltup limits, and to no longer include instrument uncertainty margins.

The proposed change will not result in physical changes to structures, systems, or components (SSCs), or to event initiators or precursors. The proposed change will not affect the ability of personnel to control RCS [Reactor Coolant System] pressure at low temperatures and, thereby, ensure the integrity of the RCPB [Reactor Coolant Pressure Boundary]. Use of ASME Code Case N–641 will be approved by the NRC through approval of a Donald C. Cook Nuclear Plant-specific exemption to requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G. Therefore, the proposed revision to the RCS pressure-temperature curve changes will have been determined in accordance with NRC accepted methodologies. These methodologies provide adequate assurance that the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes, and provide an acceptable level of protection against brittle failure. Additionally, the proposed changes will not impact the design or operation of plant systems such that previously analyzed SSCs will be more likely to fail. The initiating conditions and assumptions for accidents described in the UFSAR will remain as previously analyzed. Therefore, the proposed changes will not involve a significant increase in the probability of an accident previously evaluated.

Consequences of an Accident Previously Evaluated

The proposed change does not reduce the ability of any SSC to limit the radiological consequences of accidents described in the UFSAR. The proposed change will not alter any assumptions made in the analysis of radiological consequences of previously evaluated accidents, nor does it affect the ability to mitigate these consequences. No new or different radiological source terms will be generated as a result of the proposed change. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

The format changes will improve the appearance of the affected pages but will not affect any requirements. In summary, the probability of occurrence and the consequences of an accident previously evaluated will not be significantly increased.

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

Response: No.

The proposed change will not result in physical changes to SSCs. The proposed change will not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor will it alter the design of any plant systems. The proposed change solely involves RCS pressure-temperature limits. The types of potential accidents associated with these limits have been previously identified and evaluated. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or single failures will be introduced as a result of the proposed changes. No new or different modes of failure will be created. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed RCS pressure-temperature curves will continue to provide adequate margins of protection for the RCPB. The proposed changes have been determined, through supporting analyses, to be in accordance with the methodologies and criteria set forth in the applicable regulations, or in accordance with technically adequate alternatives. Compliance with these methodologies provides adequate margins of safety and ensures that the RCPB will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events as described in the UFSAR. The format changes will improve the appearance of the affected pages but will not affect any requirements. Therefore, the proposed change will not significantly reduce the margin of safety.

In summary, based upon the above evaluation, [Indiana Michigan Power Company] I&M has concluded that the proposed changes involve no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: September 30, 2002.

Description of amendment request: The proposed amendment requests permission to change Kewaunee Nuclear Power Plant (KNPP) Facility Operating License DRP–43 to use an upgraded computer code for design basis accident containment integrity analyses. KNPP is currently licensed to use code for Generation of Thermal-Hydraulic Information for Containment (GOTHIC) version 6.0a. The proposed amendment requests to use GOTHIC 7.0p2 (GOTHIC 7).

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.

Accident analyses affected by GOTHIC have each been evaluated and found to show good agreement between the GOTHIC 7 analysis and the current analysis of record (AOR). Safety analysis results using GOTHIC 7 are shown to satisfy all applicable design and safety analysis acceptance criteria. Since GOTHIC 7 conforms to design bases and its results are bounded by the existing safety analyses, its use within limits of the bounding accident analyses will not cause an increase in the probability or consequences of an accident previously evaluated. Adherence to safety analysis acceptance criteria prevents use of GOTHIC 7 from creating new challenges to components and systems that could adversely affect their ability to mitigate accident consequence or diminish integrity of any fission product barrier.

Thus, the requested upgrade to GOTHIC 7 with [mist diffusion layer] MDL modeling option will not increase probability or the consequences of an accident previously evaluated.

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

Upgrade to GOTHIC 7 is a change in analysis methods applied to Kewaunee [design basis accident] DBA. Analysis methods are not accident initiators. GOTHIC 7 will be applied in the same manner currently licensed and it is consistent with current plant design bases and licensed accident analysis methodologies. It does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Equipment important to safety will continue to function within design. As demonstrated by the [Numerical

Applications Inc.] NAI report, GOTHIC 7 yields a representation of expected plant response for affected design basis accidents that is more accurate but remains conservative. GOTHIC 7 predicted results for affected DBA remain bounded by the limiting analyses of record.

Thus, the requested upgrade to GOTHIC 7 does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Upgrade to GOTHIC 7 affects Kewaunee design basis [loss of coolant accident] LOCA and [main steamline break] MSLB DBA containment analyses. The results predicted by GOTHIC 7 for these DBA analyses remain within limiting design basis accidents of record. GOTHIC 7 accuracy and conservatism in this application has been verified through benchmark analyses against the current analyses of record, validated against recognized standard data, and found to be appropriate for application to Kewaunee DBA. Safety analysis acceptance criteria are satisfied and adherence to safety analysis acceptance criteria using GOTHIC 7 assures that Technical Specification limits will not be exceeded during normal operation.

Thus, upgrade to GOTHIC 7 does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: September 19, 2002.

Description of amendment request: The proposed amendment would delete Surveillance Requirement (SR) 4.6.B.2, ''Primary System Boundary—Reactor Vessel Temperature and Pressure,'' from the Monticello Technical Specifications (TSs) on the basis of the licensee's commitment to (1) relocate the current requirements to the Updated Safety Analysis Report (USAR) and (2) implement the Boiling Water Reactor Vessel and Internals Program Integrated Surveillance Program as approved by the Nuclear Regulatory Commission (NRC) in a letter dated February 1, 2002. SR 4.6.B.2 currently states: ''Test specimens representing the reactor vessel, base weld, and weld heat affected zone metal shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The material sample program shall conform to ASTM [American Society for Testing and Materials] E 185–66.'' The licensee would also make related changes to the TS Bases 3.6/4.6.

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 amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change relocates the requirement of the TS Surveillance Requirement to a Licensee controlled document and implements an integrated surveillance program that has been evaluated by the NRC staff as meeting the requirements of paragraph III.C of Appendix H to 10 CFR [Part] 50. The proposed change of relocating a TS Surveillance Requirement to the Monticello USAR and implementing an integrated surveillance program is not considered a precursor or initiator of an accident previously evaluated. The proposed change does not impact current plant operations or the design function of any structure, system or component. Consequently, the proposed change does not significantly increase the probability of any accident previously evaluated.

The proposed change provides the same assurance of Reactor Pressure Vessel integrity as has always been assured. The relocation of the TS Surveillance Requirement provides an acceptable method for implementing the integrated surveillance program which was evaluated by the NRC staff as meeting the requirements of 10 CFR [Part] 50, Appendix H, paragraph III.C. The relocation of the TS Surveillance or the implementation of an integrated surveillance program is not an input or consideration in any accident previously evaluated, thus the proposed change will not increase the probability of any such accident occurring. The proposed amendment does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor does it affect any assumptions or conditions in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly increased.

Therefore, operation of the facility in accordance with the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. No equipment interfaces are modified and no changes to any equipment function or the method of operating the equipment are being made. The proposed change, to relocate the TS Surveillance and implement an integrated surveillance program, maintains an equivalent level of RPV [reactor pressure vessel] material surveillance and does not introduce any new accident initiators. The proposed change will not change the design, configuration or operation of the plant.

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

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

The proposed amendment has been evaluated as providing an acceptable alternative to the plant-specific RPV material surveillance program that meets the requirements of the regulations for RPV material surveillance. The proposed change does not exceed or alter a design basis or safety limit. The change relocates a TS Surveillance Requirement and implements an integrated surveillance program and as such does not significantly reduce the margin of safety.

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

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

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

NRC Section Chief: L. Raghavan.

PPL Susquehanna, LLC, Docket Nos. 50–387 and 50–388, Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2), Luzerne County, Pennsylvania

Date of amendment request: September 23, 2002.

Description of amendment request: The proposed amendments would change the SSES 1 and 2 Technical Specifications (TSs) by revising limiting condition for operation (LCO) 3.6.2.3 to add a new Condition B, which permits both residual heat removal (RHR) suppression pool cooling subsystems to be inoperable for 8 hours, rather than immediately initiating a unit shutdown. By making this change, the licensee is incorporating Technical Specifications Task Force change traveler number 230 into its TSs.

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

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

The proposed change relaxes the Required Actions of [LCO 3.6.2.3] by allowing 8 hours to restore one RHR suppression pool cooling subsystem to OPERABLE status when both subsystems have been determined to be inoperable. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated. The proposed 8 hour Completion Time provides some time to restore required subsystem(s) to OPERABLE status, yet is short enough that operating an additional 8 hours is not a significant risk. Consequently, this change in Required Actions does not significantly increase the probability of occurrence of any accident previously evaluated. The Required Actions in the proposed change have been developed to provide assurance that appropriate remedial actions are taken in response to the degraded condition, considering the operability status of the RHR Suppression Pool Cooling System and the capability of minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical modification or alteration of plant equipment (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the proposed change have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Required Actions do not involve a significant reduction in a margin of safety. The proposed change has been evaluated to minimize the risk of continued operation with both RHR suppression pool cooling subsystems inoperable. The operability status of the RHR Suppression Pool Cooling System, a reasonable time for repair or replacement of required features, and the low probability of a design basis accident occurring during the repair period have been considered in the evaluation. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Richard J. Laufer.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: April 16, 2002.

Description of amendment request: The proposed change would revise the Technical Specifications to delete the primary containment isolation valves and instrumentation associated with the permanent removal of the reactor vessel head spray piping.

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 not involve a significant increase in the probability or consequences of an accident previously analyzed?

Response: No.

The proposed changes to Technical Specification Tables 3.3.2–1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 do not involve a change in structures, systems, or components that would affect the probability or consequences of any accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report.

The proposed changes involve eliminating piping and valves associated with the reactor head spray. The reactor head spray system was initially provided to cool down the steam dryer and separator during shutdown. The head spray system is not credited for the prevention or mitigation of any accident. Therefore, neither the offsite or control room radiological consequences are affected. The head spray piping removal and addition of a bolted flange on the reactor coolant pressure boundary enhances plant safety by eliminating a source of pipe whip and potential leakage. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be periodically tested in conjunction with the containment integrated leak rate test.

Therefore, as discussed above, this modification does not involve a significant increase in the probability or consequences from any accident previously analyzed.

2. Does not create the possibility of a new or different kind of accident from any accident previously analyzed?

Response: No.

The proposed changes to Technical Specification Tables 3.3.2–1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 do not involve a change in structures, systems, or components that would create a new or different kind of accident from any accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report.

The proposed change to eliminate the head spray piping and the addition of a bolted flange on the reactor coolant pressure boundary enhances plant safety by eliminating a source of pipe whip and potential leakage. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be tested in conjunction with the containment integrated leak rate test.

Therefore, as discussed above, this modification does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does not involve a significant reduction in [a] margin of safety?

Response: No.

The proposed change to delete the head spray valves from Tables 3.3.2-1, 3.3.7.4–2, 3.4.3.2–1, and 3.6.3–1 does not reduce any margin of safety as defined in the Technical Specifications or Bases. The bolted flange that will be installed on the head spray penetration will maintain the integrity of the reactor coolant pressure boundary. This flange would then be tested as part of the reactor pressure vessel hydrostatic test. In addition, the drywell penetration will be capped and welded closed. This will maintain primary containment integrity and will be tested as part of the containment integrated leak rate test.

Accordingly, based on the above, the proposed change does not involve a significant reduction in [a] margin of safety.

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

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Andersen, Acting.

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: November 7, 2001.

Description of amendment request: The proposed amendments would remove license condition 2.C.3.f from the Unit 1 operating license and license condition 2.C.4 from the Unit 2 operating license, and replace them with a commitment in Section 9.1.4.2.2.5 of the Updated Final Safety Analysis Report (UFSAR). Specifically, license conditions 2.C.3.f and 2.C.4 to FOLs NPF–2 and NPF–8, respectively, require NRC approval of the lifting devices which attach the spent fuel cask to the crane prior to use of the spent fuel cask crane for the purpose of moving spent fuel casks. Subsequent to issuance of FOLs NPF–2 and NPF–8, the NRC issued NUREG–0612, ''Control of Heavy Loads at Nuclear Power Plants,'' which endorsed the use of ANSI N14.6 for the design and inspection of special lift devises thereby eliminating the need for license conditions 2.C.3.f and 2.C.4. Accordingly, SNC proposes that license conditions 2.C.3.f and 2.C.4 be removed from FOLs NPF–2 and NPF–8, respectively, and replaced with a commitment in the FNP UFSAR to ANSI N14.6 for the design, fabrication, testing, and quality assurance requirements associated with the spent fuel cask lift device.

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 replaces license conditions 2.C.3.f and2.C.4 to FOLs NPF–2 and NPF–8, respectively, with a commitment in the FNP Updated Final Safety Analysis Report (UFSAR) to the requirements of ANSI N14.6, as clarified by NUREG–0612, for the design, fabrication, testing, maintenance, and quality assurance requirements applicable to the spent fuel cask special lift device. The proposed change does not involve a physical change to or require new or different operability requirements for plant systems, structures, or components. NUREG–0612, Control of Heavy Loads at Nuclear Power Plants, provides methods acceptable to the NRC for assuring the safe handling of heavy loads. NUREG–0612 endorses the use of ANSI N14.6 for the design, fabrication, testing, maintenance, and quality assurance requirements applicable to special lifting devices used to handle heavy loads in the proximity of safe shutdown equipment and irradiated spent fuel, thereby eliminating the need for license conditions 2.C.3.f and 2.C.4 to FOLs NPF–2 and NPF–8, respectively. Accordingly, removal of license conditions 2.C.3.f and 2.C.4 to FOLs NPF–2 and NFP–8, respectively, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change replaces license conditions 2.C.3.f and 2.C.4 from FOLs NPF–2 and NPF–8, respectively, with a commitment in the FNP UFSAR to the requirements of ANSI N14.6, as clarified by NUREG–0612, for the design, fabrication, testing, maintenance, and quality assurance requirements applicable to the spent fuel cask special lift device. The proposed change does not involve: (1) A physical change to plant systems, structures or components; or (2) require new or different operability requirements for plant systems, structures, or components. SNC's commitment to the guidance provided in ANSI N14.6, as clarified by NUREG–0612, provides assurance that the spent fuel cask special lift device, in conjunction with the use of the single-failure proof spent fuel cask crane, will preclude the possibility of a cask drop accident. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed change does not involve a physical change to the plant or impact the operability requirements of systems, structures, or components considered important to safety. As stated above, the use of ANSI N14.6, as clarified by NUREG–0612, has been endorsed by the NRC in NUREG–0612. The proposed change replaces license conditions 2.C.3.f and 2.C.4 to FOLs NPF–2 and NPF–8, respectively, with a commitment in the FNP UFSAR to the requirements of ANSI N14.6, as clarified by NUREG–0612, for the design, fabrication, testing, maintenance, and quality assurance requirements for the spent fuel cask crane special lift device. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear (SQN) Plant, Unit 1, Hamilton County, Tennessee

Date of amendment request: March 29, 2002 (TSC 02–02) as supplemented by a letter dated October 10, 2002.

Description of amendment request: The proposed amendment deletes several of the Unit 1 Technical Specification (TS) surveillance requirements (SR) contained in TS 3/4.4.5, ''Steam Generators'' (SGs), associated with the voltage-based SG alternative repair criteria (ARC). In addition the proposed changes would delete License Condition 2.C.9.d which references commitment letters associated with SG inspection activities.

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.

Tennessee Valley Authority's [TVA's] proposed TS amendment does not compromise limits associated with SG tube integrity. TVA's proposed change removes existing SG tube plugging criteria (*i.e.*, ARC) from the TS and reestablishes the standard TS criteria (40 percent through-wall criteria). This change is inherently more conservative.

The proposed revision does not alter plant equipment, test methods or operating practices. The proposed change continues to provide controls for safe operation of SQN SGs within the required limits. The proposed change does not contribute to events or assumptions associated with postulated design basis accidents (*i.e.*, SG tube rupture). The proposed change does not affect operator indicators or actions required to diagnose or mitigate a SG tube rupture accident. The proposed revisions continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

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

TVA's proposed amendment removes existing repair criteria and incorporates the more conservative TS limit for SG tube plugging (*i.e.*, plug tubes with degradation depths equal to or greater than 40 percent through-wall). This change will not give rise to new failure modes. The failure of a SG tube to maintain leakage integrity during operation is an analyzed event in the SQN Updated Final Safety Analysis Report. Accordingly, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

TVA's proposed TS amendment is conservative with respect to the margin of safety. The margin of safety is preserved through ensuring structural integrity and leakage integrity of the SG tubes.

TVA's proposed change to remove ARC from the TS does not compromise structural integrity or leakage integrity of SG tubes. The proposed change invokes the standard TS tube plugging criteria limit (40 percent through-wall criteria) which is inherently more conservative.

The proposed change does not affect the plant conditions, setpoints, or safety limits that could result in precursors to accidents or degrade accident mitigation systems. Plant system safety functions are not altered by the proposed change. Consequently, the proposed TS revisions does not reduce the margin of safety.

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

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

NRC Section Chief: Allen G. Howe.

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: September 6, 2002 (TS 00–14).

Brief description of amendments: The proposed amendments would change the Sequoyah (SQN) Units 1 and 2 Technical Specification (TS) 3/4.4.9.1, ''Pressure/Temperature [P–T] Limits, Reactor Coolant System'' and TS 3/4.4.12, ''Low Temperature Overpressure Protection [LTOP] Systems.'' The proposed amendment provides two changes to the these specifications as described below:

1. The proposed change relocates the information provided in these TSs into a pressure temperature limit report (PTLR) format in accordance with U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96–03, ''Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits.''

2. The proposed change also upgrades these TSs to the standard TS requirements for Westinghouse plants (NUREG–1431, Revision 2). In addition, the Tennessee Valley Authority (TVA) proposed a change to SQN TS 3/4.4.9.2, ''Pressurizer,'' to relocate the requirements of this TS into the SQN Technical Requirements Manual (TRM).

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

The proposed revision does not affect plant equipment, test methods or operating practices. The modification to SQN TSs is consistent with the Standard Technical Specifications for Westinghouse Plants and continues to provide controls for safe operation within the required limits. The revised specifications provide appropriate administrative controls for the RCS [reactor coolant system] P–T limits and LTOP setpoints within the PTLR for future revisions as needed. The proposed changes do not contribute to events or assumptions associated with postulated design basis accidents (DBA). The proposed revisions continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

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, test methods, or operating practices. The proposed revision to the SQN RCS P–T limits, and LTOP setpoints continues to ensure that conservative fracture toughness margins are maintained to protect against reactor pressure vessel failure and overpressure conditions. The modified P–T limits and LTOP setpoints are based on NRC approved methodology in conjunction with alternative methods provided in ASME Code Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME [American Society of Mechanical Engineers] Section XI, Division 1'' and WCAP–15315, ''Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR [pressurized water reactor] and BWR [boiling water reactor] Plants.''

The proposed changes to incorporate the PTLR format is administrative in nature and provide controls for maintaining RCS P–T limits and LTOP setpoints for future revisions as needed.

The reactor vessel P–T limits and LTOP setpoints are operational limits and are not considered to be contributors to the generation of postulated accidents. The safety functions of the associated systems remain unchanged and do not affect the assumptions of DBAs. The operational limits and setpoints continue to be governed within the TSs/PTLR. Accordingly, the proposed changes do 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.

TVA's proposed TS amendment provides revised reactor pressure vessel P–T limits and LTOP setpoints that are within the design capabilities of the RCS Safety Structures, Systems and Components (SSC) and pressure control systems. The limits are based on conservative design margins that ensure that plant operation is within the design capacity of the reactor vessel materials. Accordingly, the function of the RCS to provide a fission product barrier is not compromised.

TVA's proposed change to include revised P–T and LTOP limits does not result in a change to system design features. The proposed change does not affect plant conditions that result in precursors to accidents or cause degradation of accident mitigation systems. The plant system safety functions are not altered by the proposed change.

The proposed changes to the P–T limits and LTOP setpoints change the calculations and method from that described in the current TS Bases to one based on ASME Code Case N–640 and WCAP–15315. The effect of this change is to allow plant operation with different limits while continuing to retain conservative margins for assuring integrity of the reactor vessel and the RCS. Consequently, the proposed TS revisions do 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.

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

NRC Section Chief: Allen G. Howe.

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.

Entergy Nuclear Operations, Inc., Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: July 9, 2002.

Brief description of amendment: The proposed amendment would revise the Technical Specifications to remove the cycle-specific allowances on (1) Rod insertion limits during individual rod position indicator channel calibrations and (2) rod position indicator channel accuracy for operation at or below 50 percent power. The proposed amendment also would revise the control rod indicated misalignment limits.

Date of publication of individual notice in **Federal Register:** October 7, 2002 (67 FR 62500).

Expiration date of individual notice: November 6, 2002.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: October 1, 2002.

Brief description of amendments: The amendments revise the licensing basis as described in the Updated Final Safety Analysis Report (UFSAR) to allow lifting heavier loads with the reactor building crane during the Unit 1 refueling outage beginning in November 2002.

Date of publication of individual notice in **Federal Register:** October 4, 2002 (67 FR 62270).

Expiration date of individual notice: November 4, 2002.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

AmerGen Energy Company, LLC, et al., Docket No. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: September 11, 2001, as supplemented on June 27 and September 19, 2002.

Brief description of amendment: The amendment revised the Technical Specifications, Section 3.9, ''Refueling,'' and its corresponding bases to permit the continuation of core alterations during refueling operations with the refueling interlocks inoperable by providing alternate actions which will preserve the intended design function of the inoperable interlocks.

Date of Issuance: October 10, 2002.

Effective date: October 10, 2002, and shall be implemented within 30 days of issuance.

Amendment No.: 234.

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

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10008). The June 27 and September 19, 2002, letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 10, 2002.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of application for amendment: March 13, 2002, as supplemented May 10, August 14, September 5, September 23, and October 4, 2002.

Brief description of amendment: The amendment revises the Technical Specifications (TS) for HBRSEP2 to permit selective implementation of alternative radiological source term and modify the TS requirement for movement of irradiated fuel and performing core alterations.

Date of issuance: October 4, 2002.

Effective date: October 4, 2002.

Amendment No.: 195.

Facility Operating License No. DPR–23: Amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 30, 2002 (67 FR 21285). The May 10, August 14, September 5, September 23, and October 4, 2002, supplements contained clarifying information only and did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial application. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 31, 2002, as supplemented by letters dated June 12, June 25, July 22, September 16, and October 2, 2002.

Brief description of amendment: This amendment increases the licensed power level by approximately 1.7%, from 3,833 megawatts thermal (MWt) to 3,898 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

Date of issuance: October 10, 2002.

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

Amendment No.: 156.

Facility Operating License No. NPF–29: The amendment revises the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15622).

The June 12, June 25, July 22, September 16, and October 2, 2002, supplemental letters provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2002.

No significant hazards consideration comments received: No.

Nebraska Public Power District, Docket No. 50–298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: May 30, 2001.

Description of amendment request: The amendment revises the Cooper Nuclear Station's Technical Specifications (TS) 5.5.7, ''Ventilation Filter Testing Program (VFTP),'' reflecting a correction of an erroneous reference to American Society of Mechanical Engineers N510–1980.

Date of issuance: September 30, 2002.

Effective date: The amendment is effective on the date of issuance, to be implemented within 30 days from the date of issuance.

Amendment No.: 195.

Facility Operating License No. DPR–46: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46480). The Commission related evaluation of the amendment is contained in a Safety Evaluation dated September 30, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: June 28, 2002, as supplemented on August 15, August 16, and October 2, 2002.

Brief description of amendments: The amendments change the Salem Technical Specifications (TS) requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. TS Limiting Condition for Operation 3/4.9.3, ''Decay Time,'' is revised to allow fuel movement in the containment to commence 100 hours after the reactor has become subcritical between October 15th through May 15th. Should refueling occur between May 16th and October 14th, the current 168 hours decay time limit will remain in place. These requirements are valid through the year 2010.

Date of issuance: October 10, 2002.

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

Amendment Nos.: 251 and 232.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 30, 2002 (67 FR 55887). The August 15, August 16, and October 2, 2002, letters 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 October 10, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: July 18, 2002.

Brief description of amendments: These amendments change the Salem Technical Specifications (TSs) requirements associated with its containment spray nozzles. The frequency of TS Surveillance Requirement (SR) 4.6.2.1.d for verifying that the containment spray nozzles are unobstructed is changed from a fixed 10-year frequency to after activities that could result in nozzle blockage. In this case, PSEG will be required to evaluate the work performed to determine the impact to the containment spray system, or perform an air or smoke flow test. The applicable Bases pages are also revised to reflect this change.

Date of issuance: October 10, 2002.

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

Amendment Nos.: 252 & 233.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 20, 2002 (67 FR 53989). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: November 1, 2001, as supplemented on October 1, 2002.

Brief description of amendments: The changes modify the provisions under which equipment may be considered operable when either its normal or emergency power source is inoperable. Technical Specifications (TS) Section 3.0.5 was deleted and additional limiting conditions for operation were incorporated into electrical power systems TS 3.8.1.1, ''A.C. Sources—Operating.'' The corresponding TS Bases were modified accordingly. The proposed changes are consistent with the recommendations contained in NUREG–1431, Rev. 2, ''Standard Technical Specifications for Westinghouse Plants.''

Date of issuance: October 11, 2002.

Effective date: October 11, 2002.

Amendment Nos.: 253 and 234.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5331). The October 1, 2002 supplement was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 11, 2002.

No significant hazards consideration comments received: No.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: February 20, 2001.

Brief description of amendment: The amendment eliminates the security plan requirements from the 10 CFR Part 50 licensed site after the spent nuclear fuel has been transferred to the 10 CFR Part 72 licensed Independent Spent Fuel Storage Installation and is based in part on exemptions from specific requirements set forth in 10 CFR Part 73 and 10 CFR 50.54(p).

Date of issuance: October 10, 2002.

Effective date: October 10, 2002, to be implemented within 30 days.

Amendment No.: 131.

Facility Operating License No. DPR–54: The amendment revised the Operating License and the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 21, 2001 (66 FR 15930). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 25, 2002.

Brief description of amendment: The amendment revises Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 of the technical specifications on the reactor trip system (RTS) instrumentation. The change to SR 3.3.1.2 replaces the reference to the nuclear instrumentation system channel output by a reference to the power range channel output and deletes Note 1 to the SR. The change to SR 3.3.1.3 is editorial.

Date of issuance: October 2, 2002.

Effective date: October 2, 2002, and shall be implemented within 6 months of the date of issuance, including the incorporation of changes to the Technical Specification Bases as described in the licensee's application dated July 25, 2002.

Amendment No.: 148.

Facility Operating License No. NPF–42: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** August 20, 2002 (67 FR 53992). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 2002.

No significant hazards consideration comments received: No.

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

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 for the amendment 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.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective 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 application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Assess and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 304–415–4737 or by email to *pdr@nrc.gov*.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 29, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714,[2]

[2] ''The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 CFR 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: ''In all other circumstances, such ruling body or officer shall, in ruling on—

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

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

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

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

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

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

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

which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and electronically on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If there are problems in accessing the document, contact the PDR Reference staff at 1–800–397–4209, 301–415–4737, or by e-mail to *pdr@nrc.gov*. 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. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

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 PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of the continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov*. A copy of the petition for leave to intervene and request for hearing should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. A copy of the request for hearing and petition for leave to intervene should also be sent to 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 the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

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

Date of amendment request: September 26, 2002.

Description of amendment request: The amendments consist of a one-time change to the Dresden Updated Final Safety Analysis Report (UFSAR) to state that lifting heavy loads up to and including 116 tons is allowed prior to and during the upcoming Dresden Unit 3 refueling outage number 17.

Date of issuance: October 4, 2002.

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

Amendment No.: 196 and 189.

Facility Operating License Nos. DPR–19 and DPR–25: Amendment revises the UFSAR.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. Joliet Herald News, dated October 1, 2002. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated October 4, 2002.

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

NRC Section Chief: Anthony J. Mendiola.

Dated at Rockville, Maryland, this 18th day of October 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–27243 Filed 10–28–02; 8:45 am]

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