MD 20852. The PDR reproduction contractor will copy documents for a fee. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR reference staff by telephone at 1–800–397–4209 or (301) 415–4737, or by e-mail to *pdr@nrc.gov.*

Dated at Lisle, Illinois, this 25th day of February, 2004.

Christopher G. Miller,

Chief, Decommissioning Branch, Division of Nuclear Materials Safety, RIII.

[FR Doc. 04–5857 Filed 3–15–04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Notice

AGENCY: Nuclear Regulatory Commission.

DATES: Weeks of March 15, 22, 29, April 5, 12, 19, 2004.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of March 15, 2004

There are no meetings scheduled for the Week of March 15, 2004.

Week of March 22, 2004—Tentative

Tuesday, March 23, 2004

1:30 p.m. Briefing on Status of Office of Nuclear Security and Incident Response (NSIR) Programs, Performance, and Plans (Public Meeting) (Contact: Jack Davis, 301–415–7256).

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

2:30 p.m. Discussion of Security Issues (Closed—Ex. 1).

Wednesday, March 24, 2004

9:30 a.m. Briefing on Status of Office of Nuclear Reactor Regulation (NRR) Programs, Performance, and Plans (Public Meeting) (Contact: Mike Case, 301–415–1275).

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

Week of March 29, 2004—Tentative

There are no meetings scheduled for the Week of March 29, 2004.

Week of April 5, 2004—Tentative

There are no meetings scheduled for the Week of April 5, 2004.

Week of April 12, 2004—Tentative

Tuesday, April 13, 2004

9:30 a.m. Briefing on Status of Office of Nuclear Regulatory Research (RES) Programs, Performance, and Plans (Public Meeting) (Contact: Alan Levin, 301–415–6656).

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

Week of April 19, 2004—Tentative

There are no meetings scheduled for the Week of April 19, 2004.

* 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: Dave Gamberoni, (301) 415–1651.

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

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (201–415–1969). 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: March 11, 2004.

Dave Gamberoni,

Office of the Secretary.

[FR Doc. 04–5969 Filed 3–12–04; 9:42 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and 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, January 20, 2004, through March 4, 2004. The last biweekly notice was published on March 2, 2004 (69 FR 9857).

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. Within 60 days after the date of publication of this notice, 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.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the **Federal Register** a notice of issuance. Should the

Commission make a final No Significant Hazards Consideration Determination, any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

Within 60 days after the date of publication of this notice, 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 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed within 60 days, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: (1) The name, address and telephone number of the requestor or petitioner; (2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; (3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and (4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also set forth the specific contentions which the petitioner/requestor seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner/requestor shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner/requestor intends to rely in proving the contention at the hearing. The petitioner/requestor must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner/requestor intends to rely to establish those facts or expert opinion. The petition must include 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/requestor to relief. A petitioner/requestor who fails to satisfy 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.

If a hearing is requested, and the Commission has not made a final determination on the issue of no significant hazards consideration, 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 by: (1) First class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff; (2) courier, express mail, and expedited delivery services: Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff; (3) E-mail addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, *hearingdocket@nrc.gov;* or (4) facsimile transmission addressed to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC, Attention: Rulemakings and Adjudications Staff at (301) 415–1101, verification number is (301) 415–1966. 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 it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by email 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 requests and/or petitions and contentions will not be entertained absent a determination by the Commission or the presiding officer of the Atomic Safety and Licensing Board that the petition, request and/or the contentions should be granted based on a balancing of the factors specified in 10 CFR 2.309(a)(1)(i)–(viii).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the

NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov.*

AmerGen Energy Company, LLC, Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois

Date of amendment request: November 11, 2003.

Description of amendment request: The proposed amendment would amend Appendix A, Technical Specifications (TS), of Facility Operating License No. NPF–62 for Clinton Power Station (CPS). The proposed changes would revise several CPS TS instrument channel trip setpoint Allowable Values.

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

Response: No.

The proposed amendment implements revised Allowable Values for the following instrument functions.

- Main Steam Isolation Valve—Closure
- Anticipated Transient Without Scram Recirculation Pump Trip Reactor Steam Dome Pressure—High
- Reactor Vessel Pressure—Low (Injection Permissive)
- Reactor Vessel Water Level—Low Low Low, Level 1
- Reactor Vessel Water Level—Low Low, Level 2
- High Pressure Core Spray (HPCS) System Reactor Vessel Water Level—High, Level 8
- Reactor Core Isolation Cooling (RCIC) Storage Tank Level—Low
- HPCS System Suppression Pool Water Level—High (Pump Suction Transfer)
- Automatic Depressurization System (ADS) Initiation Permissive, Low Pressure Core Spray (LPCS) Pump Discharge Pressure—High
- ADS Initiation Permissive, Low Pressure Coolant Injection (LPCI) Pumps Discharge Pressure—High
- RCIC System Suppression Pool Water Level—High (Pump Suction Transfer)
- Main Steam Line Pressure—Low, and
- Safety Relief Valve (SRV) Relief and Low-Low Set (LLS) functions channel calibration surveillance requirement

The proposed changes do not require modification to the facility. There is no impact on the accident analysis as a result of the proposed changes to the Allowable Values. The analytical limit, which is used as input to the accident analysis, does not change. The proposed changes will be implemented through revision of the associated surveillance test procedures, where the revised Allowable Value will replace the existing value.

Derivation of the Allowable Value in accordance with Regulatory Guide 1.105, ''Instrument Setpoints,'' uses the analytical limit as a fixed starting point from which instrument uncertainties are added or subtracted, as appropriate. Calculation of the Allowable Value to plant-specific parameters provides additional confidence that protective instrumentation that passes the surveillance testing criteria will perform its design function without exceeding the associated safety analysis limit.

The revised Allowable Values for the affected equipment are not considered an initiator to any previously analyzed accident and therefore, cannot increase the probability of any previously evaluated accident. Implementation of the revised Allowable Values will ensure that the instrumentation will perform its required function to meet the accident analysis assumptions. The proposed Allowable Values will ensure that the fuel is adequately cooled, containment and drywell are isolated as required, primary containment temperature and pressure design limits are met, and overpressurization of the nuclear steam supply system is prevented following an accident or transient. The proposed changes do not increase the probability of any accident previously evaluated.

Since the proposed changes ensure the same level of protection as assumed in the accident analyses, the conclusions of the accident scenarios remain valid. As a result, no changes to radiological release parameters are involved. Therefore, the proposed changes do not increase the consequences of an accident previously evaluated.

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

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

The proposed changes do not affect the design, functional performance or operation of the facility. Similarly, they do not affect the design or operation of any structures, systems, or components involved in the mitigation of any accidents, nor do they affect the design or operation of any component in the facility such that new equipment failure modes are created. Setpoints remain the same and therefore, there is no impact on the operation of any of the associated systems.

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

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

The proposed changes do not involve a change to the plant design or operation. The proposed changes will be implemented through revisions to the associated surveillance test procedures where the revised Allowable Value replaces the existing Allowable Value. No changes to the instrument setpoints are involved. Since the availability of the systems will be maintained and since the system designs are unaffected, the proposed changes ensure the instrumentation is capable of performing their intended functions. The proposed changes do not affect the accident analyses that assume the operability of the instrumentation associated with these Allowable Values. The margins associated with the analytical limits are not impacted by the proposed Allowable Values since the analytical limits remain unchanged.

Therefore, operation of CPS in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1 (TMI–1), Dauphin County, Pennsylvania

Date of amendment request: August 6, 2003, as supplemented on February 13, 2004.

Description of amendment request: This amendment would revise the Technical Specifications (TSs) to incorporate reference to the 10 CFR 50.55a, Codes and Standards, in lieu of the existing criteria of Regulatory Guide 1.35.

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

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

Response: No.

The proposed revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates reference to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in lieu of the existing criteria of Regulatory Guide 1.35. This change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirements specified in 10 CFR 50.55a(b)(2)(vi). These regulatory requirements and the associated surveillance program ensure that the reactor building tendon prestressing system is capable of maintaining the structural integrity of the containment during operating and accident conditions. The reactor building prestressing system is not an initiator of any accident. Therefore, this change is not related to the probability of any accident previously evaluated. This change ensures that the containment tendon surveillance program addresses the appropriate regulatory criteria. This change does not result in any reduction in the effectiveness of the existing surveillance program. The tendon surveillance program will continue to ensure that the containment structure is capable of performing its intended safety function in the event of a design basis accident. Therefore, this change has no affect on the consequences of an accident previously evaluated.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations.

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

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

Response: No.

The proposed revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates reference to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in lieu of the existing criteria of Regulatory Guide 1.35. This change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirement specified in 10 CFR 50.55a(b)(2)(vi). The proposed Technical Specification change does not result in any reduction in effectiveness of the existing tendon surveillance program. The tendon surveillance program will continue to satisfy the applicable Technical Specification and regulatory required criteria, thus ensuring that the containment structure will perform its design safety function. This change has no affect on the design and operation of plant structures, systems, and components. This change does not introduce any new accident precursors and does not involve any alterations to plant configurations, which could initiate a new or different kind of accident.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations.

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

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

Response: No.

The proposed revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates reference to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in lieu of the existing criteria of Regulatory Guide 1.35. The change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirement specified in 10 CFR 50.55a(b)(2)(vi). The containment examination and inspection requirements specified in 10 CFR 50.55a(b)(2)(vi) meet the same standards as the criteria specified in Regulatory Guide 1.35. The proposed Technical Specification change does not result in any reduction in effectiveness of the existing tendon surveillance program. The tendon surveillance program will continue to satisfy the applicable Technical Specification and regulatory required criteria, thus ensuring that the containment structure will perform its design safety function in accordance with existing margins of safety for containment integrity.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations.

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: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Richard J. Laufer.

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

Date of amendments request: December 15, 2003.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination in its application dated December 15, 2003.

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

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

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

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

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: William Burton, Acting.

Carolina Power & Light Company, et al., Docket No. 50–400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: February 4, 2004.

Description of amendment request: The proposed amendment would revise the Technical Specifications Index and Technical Specifications (TS) 4.4.1.3.2, ''Reactor Coolant System Hot Shutdown Surveillance Requirements,'' and 3.4.1.4.1.b, ''Reactor Coolant System Cold Shutdown—Loops Filled Limiting Condition For Operation.'' The proposed change to the Index is an administrative update to restore consistency with other sections of the TS. The proposed change to TS 4.4.1.3.2 and TS 3.4.1.4.1.b eliminates a requirement that the wide-range instrumentation be inoperable before the narrow-range instrumentation can be used for confirmation of the minimum steam generator secondary side water level. The primary reason for this proposed change to TS 4.4.1.3.2 and TS 3.4.1.4.1.b is to provide the operational flexibility needed for a smooth transition through the applicable range of operating conditions.

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

Response: No.

There is no impact on previously evaluated accidents because the proposed amendment does not affect the capability of any structure, system, or component to perform its design function. The functional capability of the narrow range instrumentation is not impacted by the operability status of the wide range instrumentation. The existing minimum values specified by Technical Specifications for the wide range and the narrow range instrumentation conservatively incorporate the applicable uncertainties necessary to make either instrument suitable for use over the expected range of operating conditions. As a result, the proposed amendment does not affect the operating procedures and administrative controls that have the function of preventing or mitigating any [previously] evaluated accident.

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

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

Response: No.

The proposed amendment does not change the design function or operation of any structure, system, or component. The proposed amendment does not involve any physical change to plant equipment. Use of the narrow range instrumentation while the wide range instrumentation is operable does not create any new or different failure mechanisms, malfunctions, or accident initiators than those already considered in the design and licensing bases.

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

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

Response: No.

The proposed amendment does not affect the margin of safety because the existing minimum values specified by Technical Specifications for the wide range and the narrow range instrumentation are not changed. Those minimum values conservatively incorporate the applicable uncertainties necessary to make either instrument suitable for use over the expected range of operating conditions. The calculation of those uncertainties for use of the narrow range instrumentation is unaffected by the operating status of the wide range instrumentation.

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

Based on the above, [Carolina Power & Light Company] concludes that the proposed amendment involves 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 request involves no significant hazards consideration.

Attorney for licensee: Steven R. Carr, Associate General Counsel—Legal Department, Progress Energy Service Company, LLC, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Allen Howe.

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

Date of amendment request: June 3, 2003.

Description of amendment request: Pursuant to Title 10 of the Code of Federal Regulations, Section 50.90, Duke Energy Corporation requested an amendment to the McGuire Nuclear Station Facility Operating Licenses and Technical Specifications. The proposed change would add a note to Limiting Condition of Operation 3.7.11, ''Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)'', that would allow the Auxiliary Building pressure boundary to be opened intermittently under administrative control. Changes to the corresponding Bases would also be made to establish the administrative controls that are required to minimize the consequences of the open pressure boundary.

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 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) is not assumed to be an initiator of any analyzed accident. Therefore, the proposed change contained in this license amendment request has no significant impact on the probability of occurrence of any previously analyzed accident.

The ABFVES provides a means of filtering air from the area of the active emergency core cooling system (ECCS) components, thereby providing environmental control for temperature and humidity in the ECCS pump room area and the Auxiliary Building. During emergency operations, the ABFVES exhausts air from the mechanical penetration area and the ECCS pump room area and discharges it through the system filters. For cases where the Auxiliary Building pressure boundary is opened intermittently under administrative controls, appropriate compensatory measures would be required by the proposed Technical Specification to ensure the pressure boundary can be rapidly restored. Based on the compensatory measures available to the plant operators and the administrative controls required to rapidly restore an opened pressure boundary, the accident consequences do not cause a significant increase in dose above the applicable General Design Criter[i]a, Standard Review Plan, or 10 CFR [Part] 100 limits.

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

No, there are no changes being made to actual plant hardware which will result in any new accident causal mechanisms. Also, no changes are being made to the way in which the plant is being operated. Therefore, no new accident causal mechanisms will be generated.

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

No, margin of safety is related to the ability of the fission product barriers to perform their design functions during and following accident conditions. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of these barriers will not be significantly degraded by the proposed changes. When the Auxiliary Building pressure boundary is open on an intermittent basis, as permitted by the changes proposed in this license amendment request, administrative controls would be in place to ensure that the integrity of the pressure boundary could be rapidly restored. Therefore, it is expected that the plant, and the operating personnel, would maintain the ability to mitigate design basis events, and that none of the fission product barriers would be significantly affected by this change. Therefore, the proposed change is not considered to result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: October 15, 2003.

Description of amendment request: The amendments would add a new Technical Specification (TS) 3.9.7, ''Unborated Water Source isolation Valves,'' and would revise TS 3.9.2, ''Nuclear Instrumentation,'' to delete the requirement for Boron Dilution Mitigation System automatic valve actuations and makeup water pump trip during Mode 6 and to agree with the wording of NUREG–1431, ''Standard Technical Specifications Westinghouse Plants,'' Revision 2. The licensee proposed these changes to provide configuration control of the dilution valves during Mode 6 to preclude the possibility of a boron dilution event and to provide an opportunity to conduct maintenance on the volume control tank valves, refueling water storage tank valves, and their respective power supplies.

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

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

Operation of the facilities in accordance with this amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated. The BDMS [Boron Dilution Mitigation System] system is designed to mitigate the consequences of an inadvertent boron dilution event. The probability of the dilution accident will be reduced by administratively isolating potential dilution flow paths. Thus, with the proposed changes, boron dilution is not considered a credible accident during refueling.

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

Operation of the facilities in accordance with this amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms are created as a result of this proposed amendment. No changes are being made to any structure, system, or component which will introduce any new accident causal mechanisms. This amendment request does not impact any plant systems that are accident initiators and does not impact any safety analysis.

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

Operation of the facilities in accordance with this amendment would not involve a significant reduction in a margin of safety. The design criterion and margin of safety for the current BDMS is that the dilution event is terminated prior to the loss of all shutdown margin. The same criterion will be met following the isolation of dilution valves. Therefore, there is no 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: Ms. Lisa F. Vaughn, Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 18, 2004.

Description of amendment request: The proposed amendment deletes requirements from the Technical Specifications (TS) to maintain hydrogen recombiners and hydrogen and oxygen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island Nuclear Station] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated February 18, 2004.

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

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

The revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen and oxygen monitors are no longer required to mitigate design-basis accidents and, therefore, the monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen and oxygen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents. Also, as part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in RG 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The regulatory requirements for the hydrogen and oxygen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, classification of the oxygen monitors as Category 2, and removal of the hydrogen and oxygen monitors from TS will not prevent an accident management strategy through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen and oxygen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen and oxygen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen and oxygen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Category 2 oxygen monitors are adequate to verify the status of an inerted containment.

Therefore, this change does not involve a significant reduction in the margin of safety. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related oxygen monitors. Removal of hydrogen and oxygen monitoring from TS will not result in a significant reduction in their functionality, reliability, and availability.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 15, 2004.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than February 27, 2011, for Unit 2, and no later than July 13, 2009, for Unit 3.

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

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

The proposed change will revise Dresden Nuclear Power Station (DNPS) Units 2 and 3 Technical Specifications (TS) Section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than February 27, 2011, for Unit 2, and no later than July 13, 2009, for Unit 3. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor coolant system (RCS) following a design basis loss-of-coolant accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Therefore, extending this test interval on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of Type A testing provides assurance that the DNPS primary containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

The proposed change for a one-time extension of the Type A tests for DNPS Units 2 and 3 will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

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

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

DNPS Units 2 and 3 are General Electric BWR/3 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the RCS; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' The tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests do not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, EGC does not expect additional degradation, during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy 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: January 15, 2004.

Description of amendment request: Modify Technical Specification Surveillance Requirement 3.4.3.2, SR 3.5.1.10, and SR 3.6.1.6.1 to provide an alternative means for testing the main steam Electromatic relief valves and the dual function Target Rock safety/relief valves.

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

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

The proposed changes modify Technical Specifications (TS) Surveillance Requirement (SR) 3.4.3.2, SR 3.5.1.10, and SR 3.6.1.6.1 to provide an alternative means for testing the main steam line relief valves, automatic depressurization system valves, and low set relief valves. Accidents are initiated by the malfunction of plant equipment, or the catastrophic failure of plant structures, systems or components. The performance of relief valve testing is not a precursor to any accident previously evaluated and does not change the manner in which the valves are operated. The proposed testing requirements will not contribute to the failure of the relief valves nor any plant structure, system or component. Exelon Generation Company, LLC has determined that the proposed change in testing methodology provides an equivalent level of assurance that the relief valves are capable of performing their intended safety functions. Thus, the proposed changes do not affect the probability of an accident previously evaluated.

The performance of relief valve testing provides confidence that the relief valves are capable of depressurizing the reactor pressure vessel (RPV). This will protect the reactor vessel from overpressurization and allow the combination of the Low Pressure Coolant Injection and Core Spray systems to inject into the RPV as designed. The low set relief logic causes two low set relief valves to be opened at a lower pressure than the relief mode pressure setpoints and causes the low set relief valves to stay open longer, such that reopening of more than one valve is prevented on subsequent actuations. Thus, the low set relief function prevents excessive short duration relief valve cycles with valve actuation at the relief setpoint, which limits induced thrust loads on the relief valve discharge line for subsequent actuations of the relief valve. The proposed changes do not affect any function related to the safety mode of the duel function safety/relief valves. The proposed changes involve the manner in which the subject valves are tested, and have no effect on the types or amounts of radiation released or the predicted offsite does in the events of an accident. The proposed testing requirements are sufficient to provide confidence that the relief valves are capable of performing their intended safety functions. In addition, a stuck open relief valve accident is analyzed in the Updated Final Safety Analysis Report. Since the proposed testing requirements do not alter the assumptions for the stuck open relief valve accident, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed changes do not affect the assumed accident performance of the main steam relief valves, nor any plant structure, system, or component previously evaluated. The proposed changes do not install any new equipment, and installed equipment is not being operated in a new or different manner. The proposed change in test methodology will ensure that the valves remain capable of preforming their safety functions due to meeting the testing requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, with the exception of opening the valve following installation or maintenance for which a relief request has been submitted, proposing an acceptable alternative. No setpoints are being changed which would alter the dynamic response of plant equipment. Accordingly, no new failure modes are introduced.

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

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

The proposed changes will allow testing of the valve actuation electrical circuitry, including the solenoid, and mechanical actuation components, without causing the relief valve to open. The relief valves will be manually actuated prior to installation in the plant. Therefore, all modes of relief valve operation will be tested prior to entering the mode of operation requiring the valve to perform their safety functions. The proposed changes do not affect the valve setpoint or the operational criteria that directs the relief valves to be manually opened during plants transients. There are no changes proposed which alter the setpoints at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation.

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

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

Attorney for licensee: Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1, Beaver County, Pennsylvania

Date of amendment request: January 27, 2004.

Description of amendment request: The proposed change would revise Technical Specification 3.4.5 to allow repair of steam generator tubes by installation of leak limiting Alloy 800 sleeves.

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

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

No. The leak limiting Alloy 800 sleeves are designed using the applicable American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code [ASME Code] and, therefore, meet the design objectives of the original steam generator (SG) tubing. The applied stresses and fatigue usage for the sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC Regulatory Guide 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes.'' Burst testing of sleeve-tube assemblies has confirmed the analytical results and demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

The leak limiting Alloy 800 sleeve depth-based structural limit is determined using NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for the configuration of long axial cracks. An Alloy 800 sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly.

Evaluation of the repaired SG tube testing and analysis indicates no detrimental effects on the leak limiting Alloy 800 sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at Beaver Valley Power Station (BVPS) Unit [No.] 1. Corrosion testing and historical performance of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

The implementation of the proposed change has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the leak limiting Alloy 800 sleeve-tube assembly is bounded by the current SG tube rupture (SGTR) analysis described in the BVPS Unit No. 1 Updated Final Safety Analysis Report. Due to the slight reduction in the inside diameter caused by the sleeve wall thickness, primary coolant release rates through the parent tube would be slightly less than assumed for the SGTR analysis and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feedwater line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the BVPS Unit No. 1 safety analysis. The sleeve-tube assembly leakage during plant operation would be minimal and is well within the allowable Technical Specification leakage limits.

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

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

No. The leak limiting Alloy 800 sleeves are designed using the applicable ASME Code as guidance, and therefore meet the objectives of the original SG tubing. As a result, the functions of the SG will not be significantly affected by the installation of the proposed sleeve. The proposed sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing SGTR accident analysis. The continued integrity of the installed sleeve-tube assembly is periodically verified by Technical Specification requirements and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the tube wall in the leak limiting sleeve/tube assembly.

Implementation of the proposed change has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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

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

No. The repair of degraded SG tubes with leak limiting Alloy 800 sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The reduction in core cooling margin due to the addition of Alloy 800 sleeves is not significant because the cumulative effect of all repaired (sleeved) and plugged tubes will continue to be less than the currently allowed core cooling margin threshold established by the total steam generator tube plugging level. The design safety factors utilized for the sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original SG design. The sleeve and portions of the installed sleeve-tube assembly that represent the reactor coolant pressure boundary will be monitored and a sleeved tube will be plugged on detection of an imperfection in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly. Use of the previously identified design criteria and design verification testing assures that the margin to safety is not significantly different from the original SG tubes.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: January 26, 2004.

Description of amendment request: The proposed change would revise the BVPS–1 and 2 Updated Final Safety Analysis Report (UFSAR) description of the design-basis bounding limitations for the ultimate heat sink design. The proposed change would allow the design descriptions in the BVPS–1 and 2 UFSARs to credit the current Technical Specification (TS) 3.7.5.1 requirement at each unit to shut down when the Ohio River level reaches a low level below 654 feet mean sea level (msl). This UFSAR revision would preclude design consideration for design-basis accidents associated with power operation from occurring when the Ohio River level is below 654 feet msl since the units would be required to be shut down.

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

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

No. The proposed change will revise the BVPS Unit No. 1 and Unit No. 2 UFSAR description of the design basis bounding limitations for the ultimate heat sink design. FENOC's proposed change will allow the design description in each BVPS Unit's UFSAR to credit the current [TS] 3.7.5.1 requirement at each BVPS Unit to shutdown when the Ohio River level reaches a low level below 654 feet Mean Sea Level (msl). This UFSAR revision will, therefore, preclude design consideration for design bases accidents associated with power operation from occurring when the Ohio River level is below 654' msl since the plant will already be shutdown. This LAR [license amendment request] does not propose any Technical Specification changes nor any physical plant changes.

Since no physical plant changes nor any instrument setpoint changes are being requested, it [the proposed change] would not result in an increase in [the] probability of an accident previously evaluated. Since the proposed change only clarifies the limiting design basis ultimate heat sink scenario, consistent with both Units' original licensing bases, it would not result in a significant increase in the consequences of an accident previously evaluated.

In conclusion, the request to amend the UFSARs for BVPS Unit Nos. 1 and 2 to clarify the limiting design basis ultimate heat sink scenario, consistent with both Units' original licensing bases, 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 previously evaluated?

No. The proposed changes only clarif[y] the limiting design basis ultimate heat sink scenario, consistent with both Units' original licensing bases. Since this is not a change to [the] original licensing bases and the design for the River Water System, Service Water System, Intake Structure, and [the] ultimate heat sink will remain valid for all credible plant conditions, this does not induce a new mechanism that would result in a different kind of accident from those previously analyzed.

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

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

No. The proposed changes [sic] only clarif[y] the limiting design basis ultimate heat sink scenario, consistent with both Units' original licensing bases. The proposed bounding conditions bound the credible BVPS Unit 1 and Unit 2 operating conditions. The design for the River Water System, Service Water System, Intake Structure, and ultimate heat sink continue to meet General Design Criteria 2 and 44 and the recommendations of Regulatory Guide 1.27, Revision 2.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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.

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

Date of amendment request: January 28, 2004.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) to maintain hydrogen recombiners and hydrogen monitors. Licensees were generally required to implement upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide (RG) 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to combustible gas control were imposed by Order for many facilities and were added to or included in the TSs for nuclear power reactors currently licensed to operate. The revised 10 CFR 50.44, ''Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors,'' eliminated the requirements for hydrogen recombiners and relaxed safety classifications and licensee commitments to certain design and qualification criteria for hydrogen and oxygen monitors.

The NRC staff issued a notice of availability of a model no significant hazards consideration (NSHC) determination for referencing in license amendment applications in the **Federal Register** on September 25, 2003 (68 FR 55416). The licensee affirmed the applicability of the model NSHC determination in its application dated January 28, 2003.

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

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

The revised 10 CFR 50.44 no longer defines a design-basis loss-of-coolant accident (LOCA) hydrogen release, and eliminates requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that could threaten containment integrity.

With the elimination of the design-basis LOCA hydrogen release, hydrogen monitors are no longer required to mitigate design-basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. RG 1.97 Category 1, is intended for key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. The hydrogen monitors no longer meet the definition of Category 1 in RG 1.97. As part of the rulemaking to revise 10 CFR 50.44 the Commission found that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of beyond design-basis accidents.

The regulatory requirements for the hydrogen monitors can be relaxed without degrading the plant emergency response. The emergency response, in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3 and removal of the hydrogen monitors from TS will not prevent an accident management strategy through the use of the SAMGs, the emergency plan (EP), the emergency operating procedures (EOP), and site survey monitoring that support modification of emergency plan protective action recommendations (PARs).

Therefore, the elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, does not involve a significant increase in the probability or the consequences of any accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, will not result in any failure mode not previously analyzed. The hydrogen recombiner and hydrogen monitor equipment was intended to mitigate a design-basis hydrogen release. The hydrogen recombiner and hydrogen monitor equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

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

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The elimination of the hydrogen recombiner requirements and relaxation of the hydrogen monitor requirements, including removal of these requirements from TS, in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents, results in a neutral impact to the margin of safety.

The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage.

Category 3 hydrogen monitors are adequate to provide rapid assessment of current reactor core conditions and the direction of degradation while effectively responding to the event in order to mitigate the consequences of the accident. The intent of the requirements established as a result of the TMI, Unit 2 accident can be adequately met without reliance on safety-related hydrogen monitors.

Therefore, this change does not involve a significant reduction in the margin of safety. Removal of hydrogen monitoring from the TSs will not result in a significant reduction in their functionality, reliability, and availability.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

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

NRC Section Chief: Richard J. Laufer

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 3, 2004

Description of amendment request: This amendment request proposes to revise a footnote to clarify a surveillance requirement and associated bases for emergency diesel generator testing.

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

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

FPL Energy Seabrook, LLC (FPLE Seabrook) proposes to revise footnote (* * *) of Technical Specification (TS) Surveillance Requirement (SR) 4.8.1.1.2a.5 to remove the link created between actions b. and c. of TS 3.8.1.1 and the loaded surveillance testing requirements of SR 4.8.1.1.2a.6. This revision to footnote (* * *) is a change to the Technical Specifications that does not modify the physical design or operation of the plant and will not create a possibility of an accident. Strict compliance with the footnote requires paralleling the only operable EDG [emergency diesel generator] unit with the off-site grid upon entry into action statement[s] b. or c. of TS 3.8.1.1. Operation of the only operable EDG unit in this manner may increase its vulnerability for failure if power from the off-site grid is disturbed or lost. EDG unit availability for subsequent emergency demands may also be adversely affected.

The proposed change will eliminate the undesirable link that presently exists between action statement[s] b. and c. of TS 3.8.1.1 and SR 4.8.1.1.2a.6 but will maintain the primary purpose of the SR, which is to ensure that the EDG unit is capable of starting from standby conditions and attaining rated voltage and frequency. Additionally, the proposed change is consistent with the methodology used in NRC [Nuclear Regulatory Commission] NUREG–1431, Revision 3, ''Standard Technical Specifications Westinghouse Plants.'' Therefore, the proposed change does not involve a significant increase [in] the probability or consequences of an accident previously evaluated.

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

The proposed change does not affect any plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change. The proposed change to footnote (* * *) does not have the capability to initiate accidents. The proposed change will eliminate the undesirable link that presently exists between action statement[s] b. and c. of TS 3.8.1.1 and SR 4.8.1.1.2a.6. However, the proposed change will maintain the primary purpose of the SR and supporting footnote, which is to ensure that the EDG unit is capable of starting from standby conditions and attaining rated voltage and frequency. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a change in the operational limits or physical design of the plant. The proposed changes do not change the function or operation of plant equipment or affect the response of that equipment if it is called on to operate. The performance capability of the EDG units will not be affected. The proposed change will maintain the primary purpose of the SR and supporting footnote, which is to ensure that the EDG unit is capable of starting from standby conditions and attaining rated voltage and frequency. 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: M. S. Ross, Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408–0420.

Acting NRC Section Chief: Darrell J. Roberts.

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

Date of amendment request: January 29, 2004.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.4.9 Pressure Temperature (P/T) Curve figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 for Heatup/Cooldown-Core not Critical, Pressure Test and Heatup/Cooldown-Core Critical conditions.

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

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

The proposed revisions to the Cooper Nuclear Station (CNS) P/T curves are based on the recommendations in Regulatory Guide (RG) 1.99, Revision 2, and are therefore in accordance with the latest Nuclear Regulatory Commission (NRC) guidance. The evaluation for the P/T curves for 32 EFPY [Effective Full Power Years] was performed using the approved methodologies of 10 CFR [Part] 50, Appendix G. The curves generated from these methods provide guidance to ensure that the P/T limits will not be exceeded during any phase of reactor operation. Accordingly, the proposed revision to the CNS P/T curves is based on an NRC accepted means of ensuring protection against brittle reactor vessel fracture, and compliance with 10 CFR [Part] 50 Appendix G. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Based on the above, NPPD [Nebraska Public Power District] concludes that the proposed TS change to TS 3.4.9 P/T curves, figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed change updates existing P/T operating limits to correspond to the current NRC guidance. The proposed TS change provides more operating flexibility in the P/T curves for in-service leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality, with the benefits primarily in the area of pressure test being performed at a lower temperature. The proposed change does not involve a physical change to the plant, add any new equipment or any new mode of operation. These changes demonstrate compliance with the brittle fracture requirements of 10 CFR [Part] 50 Appendix G, and therefore do not create the possibility for a new or different kind of accident from any accident previously evaluated.

Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9 P/T curves, figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to the CNS P/T curves does not create a significant reduction in the margin of safety. The proposed change revises the existing CNS P/T curves to be consistent with recommendations of RG 1.99, Revision 2, the current NRC guidance given to ensure compliance with 10 CFR [Part] 50 Appendix G.

For P/T curve development ASME [American Society of Mechanical Engineers] Section Xl Code [Boiler and Pressure Vessel Code] Case N–640 uses the Kic fracture toughness curve as the lower bound for fracture toughness. P/T curves based on the Kic fracture toughness limits enhance industrial safety by expanding the P/T window in the low-temperature operating region. The potential benefits are a reduction in the duration of the pressure test and, associated increase in personnel safety, while conducting inspections in primary containment. Therefore, operational flexibility is gained while maintaining an adequate margin of safety to Reactor Pressure Vessel brittle fracture. As stated above, the development of the P/T curves to 32 EFPY was performed per the guidelines of 10 CFR [Part] 50 Appendix G, and thus, the margin of safety is not significantly reduced as the result of the proposed TS change.

Based on the above, NPPD concludes that the proposed TS change to TS 3.4.9 P/T curves, figures 3.4.9–1, 3.4.9–2, and 3.4.9–3 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. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 2, 2004.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on February 24, 2003 (68 FR 8637), on possible amendments to revise the action for one or more SDV vent or drain lines with an inoperable valve, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 15, 2003 (68 FR 18294). The licensee affirmed the applicability of the model NSHC determination in its application dated February 2, 2004

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

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

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

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

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

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 9, 2004

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of

Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, several notes or specific exceptions are revised to reflect the related changes to LCO 3.0.4, and Surveillance Requirement (SR) 4.0.4 is revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated February 9, 2004.

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

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in a Margin of Safety.

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: February 9, 2004.

Description of amendment request: The proposed amendment revises TS 5.5.7, ''Reactor Coolant Pump Flywheel Inspection Program,'' to extend the allowable inspection interval to 20 years.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 24, 2003 (68 FR 37590), on possible amendments to extend the inspection interval for reactor coolant pump (RCP) flywheels, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 22, 2003 (68 FR 60422). The licensee affirmed the applicability of the model NSHC determination in its application dated February 9, 2004.

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

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

The proposed change to the RCP flywheel examination frequency does not change the response of the plant to any accidents. The RCP will remain highly reliable and the proposed change will not result in a significant increase in the risk of plant operation. Given the extremely low failure probabilities for the RCP motor flywheel during normal and accident conditions, the extremely low probability of a loss-of-coolant accident (LOCA) with loss of offsite power (LOOP), and assuming a conditional core damage probability (CCDP) of 1.0 (complete failure of safety systems), the core damage frequency (CDF) and change in risk would still not exceed the NRC's acceptance guidelines contained in Regulatory Guide (RG) 1.174 (<1.0E–6 per year). Moreover, considering the uncertainties involved in this evaluation, the risk associated with the postulated failure of an RCP motor flywheel is significantly low. Even if all four RCP motor flywheels are considered in the bounding plant configuration case, the risk is still acceptably low.

The proposed change does not adversely affect accident initiators or precursors, nor alter the design assumptions, conditions, or configuration of the facility, or the manner in which the plant is operated and maintained; alter or prevent the ability of structures, systems, components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits; or affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed change does not increase the type or amount of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposure. The proposed change is consistent with the safety analysis assumptions and resultant consequences. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change in flywheel inspection frequency does not involve any change in the design or operation of the RCP. Nor does the change to examination frequency affect any existing accident scenarios, or create any new or different accident scenarios. Further, the change does not involve a physical alteration of the plant (*i.e.,* no new or different type of equipment will be installed) or alter the methods governing normal plant operation. In addition, the change does not impose any new or different requirements or eliminate any existing requirements, and does not alter any assumptions made in the safety analysis. The proposed change is consistent with the safety analysis assumptions and current plant operating practice. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed change will not result in plant operation in a configuration outside of the design basis. The calculated impact on risk is insignificant and meets the acceptance criteria contained in RG 1.174. There are no significant mechanisms for inservice degradation of the RCP flywheel. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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 application for amendments: March 28, 2003, as supplemented December 5, 2003.

Brief description of amendments: These amendments revise the Technical Specifications by eliminating the requirements associated with hydrogen recombiners and hydrogen monitors.

Date of issuance: March 2, 2004.

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

Amendment Nos.: 262 and 239.

Renewed Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* May 13, 2003 (68 FR 25651)

The December 5, 2003, supplemental letter provided clarifying information that did not enlarge the scope of the amendment as noticed in the original **Federal Register** notice or change the no significant hazards consideration.

The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated March 2, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 21, 2003, as supplemented February 5, 2004.

Brief Description of amendments: The amendment revised the Updated Final Safety Analysis Report (UFSAR) to describe temporary operation of the turbine building ventilation system in a once-through versus recirculation configuration during outages.

Date of issuance: February 26, 2004.

Effective date: Effective as of the date of issuance shall be implemented in accordance with 10 CFR 50.71(e).

Amendment Nos.: 230 and 258.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments approved changes to the UFSAR.

Date of initial notice in ***Federal Register***: August 5, 2003 (68 FR 46241). The February 5, 2004, supplemental letter provided clarifying information only and did not change the initial proposed no significant hazards consideration or expand the scope of the initial application. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 26, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 20, 2003, as supplemented by letters dated June 10, September 30, and October 22, 2003

Brief description of amendments: The amendments revised the Technical Specifications (TSs) to update the heatup, cooldown, criticality, and inservice test pressure and temperature limits for the reactor coolant system of each unit to a maximum of 34 Effective Full Power Years. Additionally, the amendments revise the Low Temperature Overpressure (LTOP) System TSs in order to reflect the revised pressure-temperature limits and the revised LTOP enable temperature.

Date of issuance: March 4, 2004.

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

Amendment Nos.: 212 and 206.

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

*Date of initial notice in **Federal Register**: December 23, 2003 (68 FR 74264).*

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 4, 2004.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 30, 2003, as supplemented by letter dated December 16, 2003.

Brief description of amendment: The amendment revises the control room emergency ventilation system surveillance requirements (SRs) by modifying an existing SR related to the makeup flow rate to show that it is applicable to the VSF–9 train and by adding a new makeup flow rate SR that is applicable to the 2VSF–9 train.

Date of issuance: March 2, 2004.

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

Amendment No.: 221.

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

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

The December 16, 2003, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 2, 2004.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy 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 amendments: March 28, 2003, as supplemented by letters dated October 23 and December 5, 2003.

Brief description of amendments: The amendments revise the technical specifications to reduce the main steam line low pressure primary containment isolation allowable value.

Date of issuance: February 18, 2004.

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

Amendment Nos.: 206/198, 219/213.

Facility Operating License Nos. DPR–19, DPR–25, DPR–29 and DPR–30: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* December 23, 2003 (68 FR 74265). The October 23 and December 5, 2003, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 18, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 31, 2003, as supplemented June 26, 2003.

Brief description of amendments: The amendments revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the change increases the upper limit associated with TS Table 3.3.5.1–1, ''Emergency Core Cooling System Instrumentation,'' Function 3.e, ''HPCS System Flow Rate—Low (Bypass),'' Allowable Value from less than or equal to (≤) 1704 gallons per minute (gpm) to ≤ 2194 gpm.

The change increases the Allowable Value band to account for instrumentation deadband, as-left setting tolerances and setpoint drift and to resolve historical difficulties during calibration. The current Allowable Value was initially provided in the LaSalle County Station TS during conversion to Improved Technical Specifications (ITS) format. This value was based on vendor supplied data and believed at the time to adequately account for these parameters. The upper Allowable Value limit is being increased based on historical performance data for the High Pressure Core Spray (HPCS) system flow switches. The increase in the allowed bypass flow rate does not affect the capability of the HPCS system in performing its intended safety function.

Date of issuance: March 4, 2004.

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

Amendment Nos.: 165 and 151.

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

*Date of initial notice in **Federal Register**:* May 13, 2003 (68 FR 25654). The supplement dated June 26, 2003, provided clarifying information that did not change the scope of the March 31, 2003, application nor the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 4, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 16, 2003 as supplemented January 29 and February 13, 2004.

Brief description of amendment: This amendment revised the Technical Specifications to allow a one-time extension of the steam generator tube inservice inspection interval from March 9, 2004, to March 31, 2005.

Date of issuance: February 26, 2004.

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

Amendment No.: 262.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register**:* January 6, 2004 (69 FR 695).

The supplements dated January 29 and February 13, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 26, 2004.

No significant hazards consideration comments received: No.

Nine Mile Point Nuclear Station, LLC, Docket No. 50–410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: November 20, 2003, as supplemented by letter dated February 5, 2004.

Brief description of amendment: The amendment revised Section 2.1.1.2 of the Technical Specifications to reflect the results of cycle-specific calculations performed for the upcoming Operating Cycle 10, which would employ a mixed core consisting of predominantly GE11 fuel bundles with some new GE14 fuel bundles.

Date of issuance: February 25, 2004.

Effective date: As of the date of issuance, to be implemented prior to startup from Refueling Outage 9.

Amendment No.: 112.

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

Date of initial notice in ***Federal Register****:* December 23, 2003 (68 FR 74267).

The supplemental letter of February 5, 2004, provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The staff's related evaluation of the amendment is contained in a Safety Evaluation dated February 25, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 22, 2003, as supplemented July 9, November 5, December 15, 2003, and January 30, February 9, and February 20, 2004.

Brief description of amendment: The amendment revised the Kewaunee Nuclear Power Plant operating license and technical specifications to increase the licensed rated power by 6.0 percent from 1673 megawatts thermal to 1772 megawatts thermal.

Date of issuance: February 27, 2004.

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

Amendment No.: 172.

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

Date of initial notice in ***Federal Register****:* June 10, 2003 (68 FR 34670).

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 2, 2003, as supplemented by letters dated August 8 and November 13, 2003.

Brief description of amendments: The amendments revise certain operational requirements of the Diablo Canyon Nuclear Plant Technical Specifications for the ventilation filter testing program, the control room ventilation system, the auxiliary building ventilation system, and the fuel handling building ventilation system. The amendments also incorporate a selective implementation of the alternative source term.

Date of issuance: February 27, 2004.

Effective date: February 27, 2004, and shall be implemented within 180 days from the date of issuance.

Amendment Nos.: Unit 1—163; Unit 2—165.

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

Date of initial notice in ***Federal Register****:* June 24, 2003 (68 FR 37579).

The August 8 and November 13, 2003, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 27, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 29, 2003, as supplemented January 12, 2004.

Brief description of amendment: This amendment revises the Technical Specifications (TSs) references in the Surveillance Requirement (SR) 4.0.5 and associated Basis, and Bases 3/4.4.2, 3/4.4.6, and 3/4.4.10. In the current plant TSs, the reference for inservice testing (IST) and inservice inspection (ISI) activities is the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPV Code), Section XI. The licensee proposed to reference the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and the ASME BPV Code, Section XI for IST activities and ISI activities respectively. These changes reflect the fact that the pump and valve testing requirements previously contained in Subsections IWP and IWV of the ASME BPV Code, Section XI, have been replaced by the requirements in the 1998 Edition of the ASME OM Code, 2000 Addenda, for the licensee's third 120-month IST interval. These TS changes are required to implement the IST program update in accordance with the requirements of 10 CFR.55a(f)(5)(ii). The licensee also proposed certain other language changes.

Date of issuance: February 18, 2004.

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

Amendment No.: 166.

Facility Operating License No. NPF–12: Amendment revised the TSs.

Date of initial notice in ***Federal Register****:* October 14, 2003 (68 FR 59219). The supplemental letter provided clarifying information that was within the scope of the initial notice and did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 18, 2004.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 1, 2003, as supplemented December 19, 2003.

Description of amendment request: The amendment revised the safety limit minimum critical power ratio values in Technical Specification (TS) 2.1.1.2.

Date of issuance: February 24, 2004.

Effective date: February 24, 2004.

Amendment No.: 246.

Facility Operating License No. DPR–68: Amendment revised the TSs.

Date of initial notice in ***Federal Register****:* October 28, 2003 (68 FR 61481). The December 19, 2003, letter provided clarifying information that did not change the scope of the original request or the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 24, 2004.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 8th day of March 2004.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

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