Affirmation Session (Public Meeting) (If needed).

9:30 a.m.

Briefing on License Renewal Program and Power Uprate Review Activities (Public Meeting) (Contacts: Noel Dudley, 301–415–1154, for license renewal program; Mohammed Shuaibi, 301–415–2859, for power uprate review activities).

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

2 p.m.

Meeting with Advisory Committee on Reactor Safeguards (ACRS) (Public Meeting) (Contact: John Larkins, 301–415–7360).

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

Week of July 15, 2002—Tentative

Thursday, July 18, 2002

1:55 p.m.

Affirmation Session (Public Meeting) (If needed).

Week of July 22, 2002—Tentative

There are no meetings scheduled for the Week of July 22, 2002.

Week of July 29, 2002—Tentative

There are no meetings scheduled for the Week of July 29, 2002.

Week of August 5, 2002—Tentative

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

Week of August 12, 2002—Tentative

Tuesday, August 13, 2002

9:30 a.m.

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

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

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

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

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

Dated: July 3, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–17288 Filed 7–5–02; 11:25 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from June 14, 2002 through June 27, 2002. The last biweekly notice was published on June 25, 2002 (67 FR 42814).

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

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

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

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

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

By July 25, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR

2.714,[1] which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/*. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Consumers Energy Company, Docket No. 50–155, Big Rock Point Nuclear Plant, Charlevoix, County, Michigan

Date of amendment request: June 11, 2002.

Description of amendment request: The amendment request changes the Defueled Technical Specifications by adding applicability statements to the requirements for storage and inspection of spent fuel and for the program requirements for spent fuel pool water chemistry.

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

The proposed change does not:

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

The requested license amendment involves the addition of applicability statements to the program and activity requirements for the storage and inspection of spent fuel activities and requirements and the SFP [spent fuel pool] water chemistry. These applicability statements make requirements applicable whenever irradiated fuel is stored in the SFP. Once irradiated fuel has been completely removed from the SFP and transferred to a certified dry fuel storage container under a general 10 CFR Part 72 license, these program requirements for the SFP are no longer necessary. The program requirements consist of the specification, establishment, implementation, and maintenance of fuel configuration, fuel cooling, and water chemistry for the SFP to minimize the potential effects of decay heat and corrosion.

The corresponding program requirements for fuel storage in dry containers are specified in the container's certificate of conformance and safety analysis report. The corresponding program requirements currently include:

1. Analysis of fuel assemblies to determine maximum temperatures within the fuel assemblies to the temperature at the edge of the assemblies,

2. Design of passive heat removal components to remove heat via convection, conduction, and radiation, and

3. Specifications for canister vacuum drying pressure and helium backfill pressure that would ensure that a sufficiently inert environment is produced within the canister to inhibit corrosion.

The program requirements associated with fuel storage in the SFP do not contribute to accident prevention or mitigation following the complete removal of irradiated fuel. The corresponding program features for fuel storage in dry storage containers are specified and containers are specified and controlled under other applicable license documents. These changes do not significantly increase the probability or consequences of an accident previously evaluated.

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

The requested amendment involves the addition of applicability statements that will have the effect of making a program requirement associated with the SFP inapplicable when the SFP is no longer used for irradiated fuel storage. The corresponding program requirements are adequately specified in applicable license documents. The elimination of this program requirement following complete removal of irradiated fuel from the SFP does not result in any new or different accident initiators from those already assumed in accidents previously evaluated, nor does it exacerbate any such accidents. Therefore, these changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

The safety margins produced as a result of the specification of program requirements for fuel storage in the SFP are adequately maintained in corresponding program requirements associated with fuel storage in dry storage containers. These corresponding program requirements are specified in the dry storage container's certificate of compliance and safety analysis report. Therefore, this change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's significant hazards 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: David A. Mikelonis, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Robert A. Gramm.

Dominion Nuclear Connecticut, Inc., et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: May 13, 2002.

Description of amendment request: The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 2 (MP2) and Unit No. 3 (MP3) Technical Specifications (TSs) to change selected MP2 and MP3 radiological-related TSs. These changes are due to the revision to Part 20 of Title 10 of the Code of Federal Regulations.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review is presented below:

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

These changes do not have an impact on the acceptance criteria for any design-basis accident described in the respective MP2 or MP3 Updated Final Safety Analysis Report (UFSAR).

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

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

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

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

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

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

The proposed changes have no effect on any safety analyses assumptions and therefore do not impact any margins of safety. The proposed changes do not impact any acceptance criteria for the design-basis accidents described in the respective MP2 or MP3 UFSAR and do not impact the consequences of accidents previously evaluated. Therefore, the proposed changes will not result in a reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 29, 2002.

Description of amendment request: The amendments would revise the Technical Specifications 5.5.2 to allow, on a one-time basis, extension of the interval governing the conduct of containment integrated leak rate test (ILRT) from ten to fifteen years. The amendments represent a one-time exception to the ten-year frequency of the performance-based Type A tests as delineated by Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' September 1995. The amendments will allow conduct of each respective unit's ILRT within fifteen years from the last ILRT performed for each unit.

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

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

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

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

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

First Standard

The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed extension to the Type A testing intervals cannot increase the probability of an accident previously evaluated since extension of the intervals is not a physical plant modification that could alter the probability of accident occurrence, nor is it an activity or modification by itself that could lead to equipment failure or accident initiation. The proposed extension to the Type A testing intervals does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Type B and Type C tests. It concludes that reducing the Type A testing frequency to once per twenty years leads to an imperceptible increase in risk.

Catawba and McGuire provide a high degree of assurance through testing and inspection that the containments will not degrade in a manner detectable only by Type A testing. Recent Type A tests for the Catawba and McGuire units identified containment leakage within acceptance criteria, indicating a very leak tight containment. Inspections required by the ASME Code are also performed in order to identify indications of containment degradation that could affect leak tightness. Separately, Type B and Type C testing, required by TS [Technical Specifications], identify any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the Type A test intervals will not represent a significant increase in the consequences of an accident.

Second Standard

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revisions to the Catawba and McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A test intervals does not create the possibility of a new or different type of accident since there are no physical changes being made to the plants and there are no changes to the operation of the plants that could introduce a new failure mode.

Third Standard

The proposed amendments will not involve a significant reduction in a margin of safety. The proposed revisions to the Catawba and McGuire TS add a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A test intervals will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing intervals found that a twenty-year interval resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal effect on this risk, since 95 percent of the Type A detectable leakage paths would already be detected by Type B and Type C testing. Similar proposed changes have been previously reviewed and approved by the NRC, and they are applicable to Catawba and McGuire.

Based upon the preceding discussion, Duke Energy Corporation has concluded that the proposed amendments do not involve a significant hazards consideration.

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

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

Date of amendment request: May 14, 2002.

Description of amendment request: The proposed change will revise Appendix 3B and Section 6.2.1.2 of the Updated Safety Analysis Report pertaining to the method of analysis. The proposed change will replace the current vendor THREED code for room pressure-temperature analyses due to High Energy Line Breaks (HELB) with GOTHIC (Generation of Thermal-Hydraulic Information for Containments). The proposed change will allow Entergy Operations, Inc. (EOI) to update the analysis and to evaluate additional changes to the plant.

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

1. Will the operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequence of an accident previously evaluated?

Response: The proposed change involves no increase in the probability of the accidents previously evaluated since no physical change to the plant will be made. The change of the High Energy Line Break (HELB) analysis method does not affect the probability of the analyzed event occurring.

The line break locations have not been affected and remain as originally designed.

This submittal is required due to the change of HELB analysis code from the vendor code THREED to the modern industry standard analysis code GOTHIC. This is a change in the methodology for determining the effects of the mass and energy release in the plant as a result of currently postulated events. The change in the evaluation methodology has been benchmarked and reviewed to confirm the results remain consistent with the current analysis. The changes to the model used for the additional analysis allow the use of new, more physically realistic models for Containment and Auxiliary Building pressure/temperature responses and will demonstrate continued qualification of the equipment in these buildings. Mass and energy releases for some cases have also been recalculated to credit pipe friction, which was only credited for certain cases previously.

With these new results the equipment has been reviewed and remains qualified per current programs established at RBS [River Bend Station]. Therefore, the plant will continue to function as designed and thus there will be no impact on consequences.

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

Response: No physical change to the plant will be made. The HELB locations were identified by reviewing all the possible break locations in each Auxiliary and Containment Building volume containing high-energy lines. The locations of the breaks remain the same as the previous HELB analyses. The HELB analyses have been evaluated for the current plant configuration. The new HELB analysis has been benchmarked against the previous accepted methods and found to correlate with the previous analysis. Therefore the results can be used to predict plant responses to events. The proposed change uses improved methods for mass and energy release calculation and pressure / temperature responses to determine the EQ [equipment qualification] qualification envelopes. Therefore, no new or different interaction would be created.

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

Response: The operation of the facility in accordance with the proposed changes will not involve a significant reduction in a margin of safety.

The GOTHIC code has been successfully benchmarked versus the vendor THREED code, which was used in the original design calculations. The HELB analysis results with the benchmarking GOTHIC model are consistent with the THREED results. Therefore, the use of GOTHIC code will not involve a reduction in an identified margin of safety. Given that GOTHIC code is an improved methodology and it has been extensively qualified against the solved analytical problems and testing results, the use of GOTHIC code will produce more accurate pressure/temperature responses for the HELB analyses. The use of the GOTHIC code has been approved for pressure/ temperature responses analysis at various other plants including Joseph M. Farley Nuclear Plant, Units 1 and 2, and Waterford [Steam Electric Station, Unit] 3.

The results with the revised methods will be used to show that safety equipment meets the EQ requirements. The peak temperatures and pressures in the HELB GOTHIC benchmark model are within the existing EDC [environmental design criteria] envelopes. Therefore, the pressure/ temperature responses from the HELB benchmark analyses have no impact on the equipment qualification.

The methodology in the original design calculations is very conservative. The mass and energy releases without crediting friction introduce excessive amount of high-energy fluid into the break rooms, which is unrealistic. Some HELB calculations have credited both the frictional flows and the additional zone to eliminate excessive conservatism in the pressure/temperature responses. There is no reduction in a margin of safety and the design room differential pressure limits continue to be [met].

The use of this method by EOI RBS is consistent with the guidance given in NRC [U.S. Nuclear Regulatory Commission] Generic Letter 83–11 and Supplement 1, addressing the performance of safety analyses by licensees. EOI has implemented this guidance for the GOTHIC methodology consistent with the intended application. The GOTHIC methodology has been verified and validated by the software vendor. In addition, this methodology is controlled by EOI procedures and under the EOI quality assurance program. This includes EOI and RBS specific verification and validation of this application of GOTHIC and review of the calculations performed.

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

Attorney for licensee: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc. (ENO), Docket No. 50–003, Indian Point Nuclear Generating Station, Unit 1, Buchanan, New York

Date of application for amendment: May 30, 2002.

Description of amendment request: The proposed changes will modify the Indian Point Generating Station, Unit 1 (IP1), Technical Specifications (TSs) and Provisional Operating License No. DPR–5. IP1 is completely enclosed within the protected area for Indian Point Nuclear Generating Station, Unit 2 (IP2). IP1 depends on the IP2 TSs and processes for the implementation of certain regulatory requirements. The requested changes will simplify the IP1 TSs to facilitate the IP2 transition to the Improved TSs. The IP1 TSs will be reformatted, reordered and repaginated for consistency and clarity. ENO also proposes that certain changes supersede requirements of the ''Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility'' [2] (the Order) to ensure compliance with the current requirements of 10 CFR Part 50.59, ''Changes, tests, and experiments.'' and 10 CFR Part 50.82, ''Termination of license,'' for evaluating whether changes can be made to IP1 without NRC approval.

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

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

The NSB [Nuclear Services Building] sewage effluent line radiation monitor is not required to function to mitigate any postulated accident. The design or operation of the radiation monitor on the existing sewage effluent discharge line will not be changed by deleting operability and surveillance requirements for the NSB sewage effluent radiation monitor from the IP1 TS. The nuclear services building sewage effluent line is neither an accident initiator nor mitigator.

The other proposed changes do not result in a change to the design or operation of any plant structure, system or component. Therefore any assumptions of the operability or performance of any structure, system or component in accident evaluations are unchanged.

The proposed fire protection TS 2.11 involves deleting requirements from the IP1 TS that are solely applicable to IP2. Any assumptions of the operability or performance of any structure, system or component in IP2 accident evaluations, including the Fire Plan, are unchanged. Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed TS change involves the deletion of operability and surveillance requirements for radioactive effluent monitoring of the NSB sewage effluent from the IP1 TS. The proposed TS changes do not

[2] U.S. Nuclear Regulatory Commission (NRC) letter to Consolidated Edison, ''Order to Authorize Decommissioning and Amendment No. 45 to License No. DPR–5 for Indian Point Unit 1 (TAC No. M59664),'' dated January 31, 1996.

affect the design or operation of any plant structure, system, or component.

Therefore, the proposed changes do 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?

This change to TS 1.0 does not affect a design function for or the operation of any plant structure, system, or component. The change does not affect the method of ENO's compliance with any regulation.

The proposed TS change involving IP1 TS 2.11 statement governs the protection of IP2 safe shutdown systems from fire. Effective protection of IP2 safe shutdown systems from fire is mandated by IP2 License Condition 2.K. The effectiveness of ENO compliance with IP2 License Condition 2.K is not affected by this change. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

The proposed changes to TS sections 3.1 and 3.2 involve eliminating the duplication of requirements in the IP1 TS and incorporating the requirements by reference to the IP2 TS. A single ENO organization operates both IP1 and IP2. The effective organizational requirements to ensure compliance with all ENO IP1 and IP2 site requirements are mandated by the IP2 TS. The effectiveness of ENO's safety management of the Indian Point site is not affected by this change. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

The proposed TS change to sections 4.1 and 5.2 involves eliminating the reference in the IP1 TS to the specific applicable section number of the IP2 TS. A single organization operates both IP1 and IP2. The applicable IP2 TS is obvious by the activity title. The effectiveness of ENO's safety management of the Indian Point site is not affected by this change. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

Effective compliance with the 10CFR20 requirements for radiation protection and monitoring radioactive effluent releases is mandated by other IP1 and IP2 TS and license provisions. The effectiveness of ENO compliance with 10CFR20 requirements is not adversely affected by the elimination of TS requirements for the radiation protection plan and radioactive effluent monitoring on the nuclear services building sewage effluent line.

The proposed TS change involves requirements for the site Meteorological Monitoring and Radiological Environmental Monitoring programs. However, IP2 TS provisions mandate effective compliance for meteorological and radiological environmental monitoring. The effectiveness of ENO compliance with 10CFR50.47, 10CFR100, and 10CFR20 requirements is not adversely affected by this change. In addition, this change does not affect a design function or the operation of any plant structure, system, or component. IP2 TS provisions mandate effective compliance with requirements for radiation protection. The effectiveness of ENO's compliance with 10 CFR 20 is not adversely affected by this change or the change to the section for sealed sources. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

The proposed TS change involves the location of routine and event reporting requirements. However, other IP2 TS provisions mandate effective compliance with reporting requirements. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

The effectiveness of ENO's compliance with 10CFR50.59 is not adversely affected by the clarification and relocation of the applicability of the FSAR [Final Safety Analysis Report]. In addition, this change does not affect a design function or the operation of any plant structure, system, or component.

Therefore, the change does not result in a change to any of the safety analyses or any margin of safety.

ENO also requests that the expiration date of IP1 Provisional Operating License No. DPR–5 be changed from ''midnight, October 14, 2002,'' to ''midnight, September 28, 2013,'' the current expiration date for Facility Operating License No. DPR–26 for IP2.

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

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

In its Safety Evaluation and Environmental Assessment for its January 31, 1996, Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility, the NRC evaluated the acceptability of the possession-only license and safety issues related to SAFSTOR of Indian Point Nuclear Generating Unit No. 1 until September 28, 2013. The requested change does not involve any activity that could change the assumptions of the prior Safety Evaluation and Environmental Assessment.

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

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

In its Safety Evaluation and Environmental Assessment for its January 31, 1996, Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility, the NRC evaluated the acceptability of the possession-only license and safety issues related to SAFSTOR of Indian Point Nuclear Generating Unit No. 1 until September 28, 2013. The requested change does not involve any activity that could change the assumptions of the prior Safety Evaluation and Environmental Assessment.

Therefore, the proposed amendment does 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?

In its Safety Evaluation and Environmental Assessment for its January 31, 1996, Order Approving Decommissioning Plan and Authorizing Decommissioning of Facility, the NRC evaluated the acceptability of the possession-only license and safety issues related to SAFSTOR of Indian Point Nuclear Generating Unit No. 1 until September 28, 2013. The requested change does not involve any activity that could change the assumptions of the prior Safety Evaluation and Environmental Assessment.

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

The NRC staff has reviewed the licensee's analyses 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 Fulton, Assistant General Consul, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Robert A. Gramm.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: May 30, 2002.

Description of amendment request: The proposed amendment would revise the Facility Operating License and Technical Specifications (TSs) to increase the licensed core thermal power level to 3067.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3025 MWt. The proposed power uprate involves the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, ''ECCS Evaluation Models,'' to Part 50 of Title 10 of the Code of Federal Regulations. In addition, changes would be made in TS Sections 2.2, 3.3, 3.4, 3.7, and the applicable TS Bases would be revised to account for the change in power level.

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 evaluations associated with this proposed change to core power level have demonstrated that all applicable acceptance criteria for plant systems, components, and analyses (including the Final Safety Analysis Report Chapter 14 safety analyses) will continue to be met for the proposed 1.4% increase in licensed core thermal power for IP3 [Indian Point Unit 3]. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or the operational performance of any potentially affected system, component or analysis. Therefore, the probability of an accident previously evaluated is not affected by this change. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its intended safety function. Further, the radiological dose evaluations in support of this power uprate effort show that the current FSAR [Final Safety Analysis Report] Chapter 14 radiological analyses are unaffected, and that the current dose analyses of record bound plant operation with the subject increase in licensed core thermal power level.

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 evaluations of this proposed amendment show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for the proposed 1.4% power increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analyses. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its safety function. Furthermore, the conditions associated with the subject increase in core thermal power will neither cause initiation of any accident, nor create any new credible limiting single failure. The power uprate does not result in changing the status of events previously deemed to be non-credible being made credible. Additionally, no new operating modes are proposed for the plant as a result of this requested change.

Therefore, the subject increase in core thermal power level will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The evaluations associated with this proposed change show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for this proposed 1.4% increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analysis. The subject power uprate will not adversely affect the ability of any safety-related system to meet its intended safety function. For example, most IP3 analyses already add a 2% uncertainty allowance to the nominal power level to account solely for power measurement uncertainty. These analyses have not been revised for the 1.4% uprate power level conditions because the sum of increased core power level (1.4%) and the improved power measurement accuracy (uncertainty less than 0.6%) is already bounded by the currently analyzed 2% uncertainty allowance.

Therefore, the subject increase in core thermal power will not involve a reduction in [a] margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 3, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.4.9, ''Pressurizer,'' to increase the pressurizer water level limit when the plant is in Mode 3 (Hot Standby). The current pressurizer water level limit is applicable for Modes 1, 2, and 3, and will remain unchanged for Modes 1 and 2. The proposed amendment would also revise TS 3.8.4, ''DC Sources—Operating,'' to remove the notes that refer to the one-time amendment allowing the online replacement of station batteries 31 and 32. The notes are no longer applicable since the batteries have been replaced.

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.

Pressurizer water level is an assumed initial condition for certain accident analyses. Plant initial conditions are not accident initiators and do not have an effect on the probability of the accident occurring. The proposed change only revises the specified limit on water level in the pressurizer, so that this change would not affect accident probability.

The specific accidents for which pressurizer water level is an assumed initial condition are a loss of load and a loss of normal feedwater. The limiting accident analysis results occur at full power conditions when the available core thermal power is maximized. The proposed change does not affect the specified pressurizer level limit at any power level from zero to full power. That is, the pressurizer level limit is not being changed in Modes 1 and 2. The proposed change does revise the specified pressurizer water level limit in Mode 3 (Hot Standby) but this does not affect accident analysis results because the limiting analyses will remain those that are postulated to occur in Mode 1 with the plant at full power.

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 change does not involve physical changes to existing plant equipment or the installation of any new equipment. The design of the pressurizer, the pressurizer level control system and the pressurizer safety valves is not being changed and the ability of these systems, structures, and components to perform their design or safety functions is not being affected. The proposed change revises the specified limit on pressurizer water level in Mode 3 (Hot Standby) to allow operators greater flexibility in performing a plant cooldown. The method used in performing the plant cooldown is not being changed. This proposed change does not create new failure modes or malfunctions of plant equipment nor is there a new credible failure mechanism.

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?

Response: No.

Pressurizer level is an initial condition assumed in certain accident analyses involving an insurge in the pressurizer and an increasing reactor coolant system (RCS) pressure. These analyses demonstrate that the design pressure for the RCS is not exceeded for the limiting analyses based on the plant at full power. The proposed change does not affect the existing Technical

Specification requirement for Mode 1 (Power Operation) or Mode 2 (Plant Startup) and therefore does not affect the assumptions or results of these accident analyses. The margin for RCS design pressure demonstrated by these analysis results is not being reduced. The proposed change only applies to the pressurizer level limit in Mode 3 (Hot Standby) when there is substantially lower thermal energy available to cause rapid expansion of reactor coolant and an insurge to the pressurizer. Protection of the RCS pressure boundary is still maintained by the pressurizer safety valves, which are not being modified by the proposed change in pressurizer water level.

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: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: June 5, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to implement the alternate source term methodology for the fuel-handling accident analysis. Specifically, the proposed amendment would revise TS 3.9.3, ''Containment Penetrations,'' to: (1) Permit the equipment hatch opening and the personnel air lock doors to be capable of being closed during movement of irradiated fuel, (2) allow use of administrative controls for unisolating containment penetrations during movement of irradiated fuel, (3) delete the containment purge and containment pressure relief requirements and associated surveillances with the reactor subcritical for less than 550 hours, and (4) eliminate the TS applicability ''during core alterations.'' In this regard, the proposed amendment would adopt TS Task Force (TSTF) Standard TS Change Travelers TSTF–68, ''Containment Personnel Airlock Doors Open During Fuel Movement,'' TSTF–312, ''Administratively Control Containment Penetrations,'' and, in part, TSTF–51, ''Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations.'' The proposed amendment would also relocate the requirements in TS 3.7.13, ''Fuel Storage Building Emergency Ventilation System,'' and TS 3.3.8, ''Fuel Storage Building Emergency Ventilation System Actuation Instrumentation,'' to the licensee-controlled Technical Requirements Manual.

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 change involves the reanalysis of a fuel handing accident (FHA) in containment and in the fuel storage building. The new analysis, based on the Alternate Source Term (AST) in accordance with 10 CFR [Code of Federal Regulations] 50.67, will replace the existing analysis based on methodologies and acceptance criteria in place when Indian Point 3 was originally licensed. As a result of the new analysis, changes to the Technical Specifications are proposed which take credit for the new analysis results.

The proposed changes to the technical specifications modify requirements regarding containment closure during movement of irradiated fuel assemblies in containment and relocate requirements for the fuel storage building emergency ventilation system from the technical specifications to a licensee controlled document. The proposed changes do not involve physical modifications to plant equipment and do not change the operational methods or procedures used for moving irradiated fuel assemblies. As such, there are no accident initiators affected by the proposed amendment. The revised requirements apply only when the plant is in a refueling condition (Mode 6), and specifically only when irradiated fuel is being moved. Previously evaluated accidents with the plant in other conditions ranging from cold shutdown (Mode 5) through power operation (Mode 1) are not affected. The AST methodology is used to evaluate a[n] FHA that is postulated to occur during fuel movement activities in the containment building and the fuel storage building. The analysis follows the guidance of the NRC Regulatory Guide 1.183 and uses the acceptance criteria of the NRC Standard Review Plan (NUREG 0800) for offsite doses and General Design Criteria 19 for control room personnel. The analysis demonstrates that the dose consequences meet regulatory acceptance criteria. The accident analysis conservatively assumes that the containment building and the fuel storage building, including ventilation filtration systems for those building[s] does not diminish or delay the assumed fission product release. The analysis does take credit for, and technical specifications enforce, the presence of 23 feet of water over the irradiated fuel while fuel movement activities are being performed. The analysis also takes credit for, and the technical specification bases enforce a fuel decay time of at least 84 hours. In addition, administrative controls are put in place to provide for closure of containment openings in the event of a[n] FHA. Use of an alternate analysis method does not affect fuel parameters or the equipment used to handle the fuel. The proposed changes to the technical specifications reflect assumptions made in the analysis.

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 amendment involves the use of an alternate analysis methodology for the evaluation of the dose consequences from a[n] FHA that is postulated to occur in either the containment building or the fuel storage building (FSB). The analysis demonstrates that containment closure conditions and operation of the containment purge filtration system are not required to maintain dose consequence within regulatory limits following a postulated FHA in containment. Therefore the new analysis supports proposed changes to requirements for containment closure during movement of irradiated fuel assemblies in containment. The analysis results also demonstrate that operation of the fuel storage building emergency ventilation system is not required to maintain dose consequences within regulatory limits following a postulated FHA in the FSB. The containment closure components (e.g., equipment hatch, personnel airlock doors, and various containment penetrations) and filtration systems are not accident initiators. The proposed changes do not involve the addition of new systems or components nor do they involve the modification of existing plant systems. The proposed changes do not affect the way in which a[n] FHA is postulated to occur.

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?

Response: No.

The existing dose analysis methodology and assumptions demonstrates that the dose consequences of a[n] FHA are within regulatory limits for whole body and thyroid doses as established in 10 CFR 100. The alternate dose analysis methodology and assumptions also demonstrates that the dose consequences of a[n] FHA are within regulatory limits. The limits applicable to the alternate analysis are established in 10 CFR 50.67 in conjunction with the TEDE (total effective dose equivalent) acceptance directed in Regulatory Guide 1.183. The acceptance criteria for both dose analysis methods have been developed for the purpose of evaluating design basis accidents to demonstrate adequate protection of public health and safety. An acceptable margin of safety is inherent in both types of acceptance criteria.

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: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc., 440 Hamilton Avenue, White Plains, NY 10601.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 7, 2002.

Description of amendment request: The proposed amendment would change the requirements associated with handling irradiated fuel and performing core alterations. Specifically, the changes would eliminate operability requirements for secondary containment when handling recently irradiated fuel and during core alterations. The amendment would also revise the requirements associated with equipment whose performance is not credited in the new calculations.

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

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

Response: No.

The proposed TS [Technical Specifications] changes do not modify the design or operation of equipment used to move spent fuel or to perform core alterations. Because the equipment affected by the change is not an initiator to any previously analyzed accident, the proposed change cannot increase the probability of any previously analyzed accident.

The conservative re-analysis of the fuel handling accident concludes that radiological consequences are within the acceptance criteria in Regulatory Guide 1.183 and 10 CFR 50.67. The results of the core alteration events, other than the fuel handling accident, remain unchanged from the original design-basis, which showed that these events do not result in fuel cladding damage or radioactive release. The radiological analysis uses the same FHA [fuel handling accident] source activity previously accepted in the design-basis FHA analysis. The same source activity is used with the guidance in the Regulatory Guide 1.183, Appendix B and the passive release/transport path, which does not take the dose mitigation credit of engineered safeguards including secondary containment and CREVAS [Control Room Emergency Ventilation] Systems.

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

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

Response: No.

The proposed post-FHA activity transport path is passive in nature and it does not take the credit of dose mitigation functions previously credited in the design-basis FHA analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant.

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

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

Response: No.

The proposed changes revise the FitzPatrick TS to establish operational conditions where specific activities represent situations during which significant radioactive releases can be postulated. These new operational conditions are consistent with the proposed design-basis accident analysis and are established such that the radiological consequences are less than the regulatory allowable limits. Safety margins and analytical conservatisms are retained to ensure that the analysis adequately bounds all postulated event scenarios. The selected assumptions and release models provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensates for large uncertainties in facility parameters, accident progression, radioactive material transport and atmospheric dispersion. The proposed TS applicability statements continue to ensure that the TEDE [Total Effective Dose Equivalent] at the control room and the exclusion area and low population zone boundaries are below the corresponding regulatory allowable limits in 10 CFR 50.67(b)(2).

Therefore, these 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. David E. Blabey, 1633 Broadway, New York, New York 10019.

NRC Section Chief: Richard J. Laufer.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Dockets Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York County, Pennsylvania

Date of application for amendments: May 24, 2002.

Description of amendment request: Exelon Generation Company, LLC, the licensee, is proposing changes to the Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS), Operating Licenses and Technical Specifications associated with an increase in the licensed power level. The changes involve a proposed 1.62 percent increase in the licensed reactor core thermal power level (an increase in reactor power level from 3,458 megawatts thermal to 3,514 megawatts thermal). These changes result from increased accuracy of the feedwater flow and temperature measurements to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation. This results in a more accurate determination of reactor core thermal power level. The basis for this change is consistent with the revision, issued in June 2000, to Appendix K to Part 50 of Title 10 of the Code of Federal Regulations, allowing operating reactor licensees to use an uncertainty factor of less than 2 percent of rated reactor thermal power in analyses of postulated design-basis loss-of-coolant accidents.

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 comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of all components and systems that could be affected by this change. Evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive housings, piping and supports, recirculation pumps, etc.) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the [Nuclear Steam Supply System] NSSS systems will still perform their intended design functions during normal and accident conditions. The balance of plant [(BOP)] systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level.

Because the integrity of the plant will not be affected by operation at the uprated condition, it is concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the core thermal power uncertainty measurement allows most of the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 3514 megawatts thermal (MWt). Other analyses performed at a nominal power level have either been evaluated or re-performed for the 1.62% increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.62% uprate conditions. As such, all PBAPS Updated Final Safety Analysis Report (UFSAR) Chapter 14 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.62% uprated condition.

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

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

Response: No. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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?

Response: No. Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC, or that are in compliance with regulatory review guidance and standards.

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

NRC Section Chief: James W. Clifford.

Exelon Generation Company, LLC, Docket No. 50–254, Quad Cities Nuclear Power Station, Unit 1, Rock Island County, Illinois

Date of amendment request: May 30, 2002.

Description of amendment request: The proposed change revises the safety limit minimum critical power ratio for Unit 1 Cycle 18 for two loop operation and single loop operation.

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the safety limit for the minimum critical power ratio (SLMCPR) for Quad Cities Nuclear Power Station (QCNPS), Unit 1, Cycle 18 such that the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences.

Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during any transients or anticipated operational occurrences. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criteria (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and anticipated operational occurrences) is met. Since the operability of plant systems designed to mitigate any consequences of accidents has not changed, the consequences of an accident previously evaluated are not expected to increase.

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

Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated? Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. The proposed change does not involve any modifications of the plant configuration or allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for QCNPS, Unit 1, Cycle 18.

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

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

The value of the proposed SLMCPR provides a margin of safety by ensuring that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation.

This will ensure that the fuel design safety criteria (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation as well as anticipated operational occurrences) are met.

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 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.

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

Date of amendment request: June 13, 2002.

Description of amendment request: The amendment would revise the Improved Technical Specifications (ITS) 3.3.8 and associated bases, ''Emergency Diesel Generator (EDG) Loss of Power Start (LOPS),'' by changing the completion time for required action D.2 from 12 to 36 hours. The amendment also corrects a typographical error in ITS 3.3.8 and clarifies the discussion in Bases Section B 3.3.8 for Actions D.1 and D.2 to recognize the applicability of ITS 3.3.8 in MODES 5 and 6.

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

(1) Does not involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed license amendment revises the Required Time to place the plant in MODE 5 if an inoperable loss of voltage Function for the emergency diesel generator (EDG) loss of power start (LOPS) cannot be restored to OPERABLE status, corrects a typographical error in the Section Number of ITS 3.3.8, and clarifies the wording of ITS Bases Section B 3.3.8 for Action D.1 and D.2 regarding the applicability of the specification during MODES 5 and 6.

The EDG LOPS is intended to protect engineered safeguards equipment from damage due to sustained undervoltage conditions, and to ensure rapid restoration of power to the engineered safeguards electrical buses in the event of a loss of offsite power. The EDG LOPS is not an initiator of any design basis accident. The design functions of the EDG LOPS and the initial conditions for accidents that require an EDG LOPS will not be affected by the change. Therefore, the change will not increase the probability or consequences of an accident previously evaluated.

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

The proposed amendment involves no changes to the design functions or operation of the EDG LOPS. Editorial corrections, clarification of the wording in Bases Section B 3.3.8, or changing the Required Completion Time for placing the plant in MODE 5 when an inoperable loss of voltage function cannot be restored will not introduce any new failure mechanisms, malfunctions or accident initiators. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does not involve a significant reduction in the margin of safety.

The proposed change corrects a typographical error, clarifies the wording of Bases Section B 3.3.8 for Actions D.1 and D.2, and revises the required Completion Time to place the plant in MODE 5. The revised Completion Time will allow the plant to be shutdown in an orderly fashion without challenging plant systems or plant cooldown limits. The proposed change does not change the design or operation of the EDG LOPS, and does not impact the ability of the EDG LOPS to perform its design functions. Thus, the proposed amendment will not result in a 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Acting Section Chief: Kahtan N. Jabbour.

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

Date of amendment request: June 7, 2002.

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

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the NSHC determination in its application dated June 7, 2002. The NSHC determination is restated below.

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

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

The PASS was originally designed to perform many sampling and analysis functions. These functions were designed and intended to be used in post accident situations and were put into place as a result of the TMI–2 accident. The specific intent of the PASS was to provide a system that has the capability to obtain and analyze samples of plant fluids containing potentially high levels of radioactivity, without exceeding plant personnel radiation exposure limits. Analytical results of these samples would be used largely for verification purposes in aiding the plant staff in assessing the extent of core damage and subsequent offsite radiological dose projections. The system was not intended to and does not serve a function for preventing accidents and its elimination would not affect the probability of accidents previously evaluated.

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

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

Therefore, the elimination of PASS requirements from Technical Specifications (TS) (and other elements of the licensing bases) does not involve a significant increase in the consequences of any accident previously evaluated.

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

The elimination of PASS related requirements will not result in any failure mode not previously analyzed. The PASS was intended to allow for verification of the extent of reactor core damage and also to provide an input to offsite dose projection calculations. The PASS is not considered an accident precursor, nor does its existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radioisotopes within the containment building.

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

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

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

Therefore, this change does not involve a significant reduction in [a] margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: May 29, 2002.

Description of amendment request: The proposed amendment would revise TS 3.8.1, ''AC Sources—Operating,'' to allow portions of Surveillance Requirement (SR) 3.8.1.5 to be performed with the units in Mode 1, 2, 3 or 4. This proposed amendment is consistent with changes made to NUREG–1431, Standard Technical Specifications, Westinghouse Plants, by Technical Specification Task Force (TSTF) Traveler, TSTF–283, Revision 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. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The standby emergency power sources are primarily a support system for systems required to be operable for accident mitigation. SR 3.8.1.5 demonstrates the standby emergency power source operation, during a loss of offsite power actuation test signal in conjunction with an Engineering Safeguards Feature (ESF) actuation signal. The proposed amendment only changes the allowed operating Modes in which portions of this surveillance may be performed. Performing portions of the surveillance in Mode 1, 2, 3, or 4 will require an assessment to determine that plant safety is maintained or will be enhanced.

Therefore, the consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

2. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. These changes do not introduce any new or different normal operation or accident initiators. Performing the surveillance in Mode 1, 2, 3, or 4 will require an assessment to determine that plant safety is maintained or will be enhanced.

Equipment important to safety will continue to operate as designed. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendment will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The standby emergency power sources are primarily a support system for systems required to be operable for accident mitigation. SR 3.8.1.5 demonstrates the standby emergency power source operation, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. Performing the surveillance in Mode 1, 2, 3, or 4 will require an assessment to determine that plant safety is maintained or will be enhanced. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed amendment will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other structures, systems or components (SSCs) important to safety. Therefore, allowing a portion of the surveillance to be performed in Mode 1, 2, 3, or 4, will not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: L. Raghavan.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 22, 2002.

Description of amendment request: The proposed amendment revises Technical Specifications (TSs) 3/4.3.5, allowing the automatic operation of the atmospheric steam relief valves during Mode 2 to maintain secondary side pressure at or below an indicated steam generator pressure of 1225 psig during startup and shutdown of the reactors.

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 change only provides another method of controlling the SG PORVs [steam generator power-operated relief valves] under specified operating conditions. The operating conditions in Specification 3/4.3.5 remain unchanged. No change is required to plant design since the proposed method of control is already part of the plant's configuration. The proposed method of control is the same method of control normally required by the specification in Modes 1 and 2. The proposed method of control will not impact the accident analysis assumptions or results. 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 method of controlling the SG PORVs is the same method that these valves are controlled in Modes 1 and 2 by the specification under normal conditions. The proposed change will allow the setpoint of these valves to be adjusted to support startup and shutdown activities. The adjustment of the setpoint is restricted so that the accident analysis is not impacted. No change to the design of the valves or plant configuration is required to implement the proposed change. 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?

Response: No.

The proposed change that will allow for an additional method of controlling the SG PORVs during startup and shutdown activities is consistent with the operating restrictions for the current method of valve control. The accident analysis assumptions and results will remain unaffected. 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis, & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

STP Nuclear Operating Company, Docket Nos. 50–498 and 50–499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 23, 2002.

Description of amendment request: The proposed amendment revises the near-end of life (EOL) Moderator Temperature Coefficient (MTC) Surveillance Requirements by placing a set of conditions on core operation.

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

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

Response: No.

The probability or consequences of accidents previously evaluated in the UFSAR [updated final safety analysis report] are unaffected by this proposed change because there is no change to any equipment response or accident mitigation scenario. There are no additional challenges to fission product barrier integrity. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related system. 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?

Response: No.

The margin of safety associated with the acceptance criteria of any accident is unchanged. The proposed change will have no affect on the availability, operability, or performance of the safety-related systems and components. A change to a surveillance requirement is proposed, but the limiting conditions for operation required by the Technical Specifications are not changed.

The Technical Specifications Bases are founded in part on the ability of the regulatory criteria to be satisfied assuming the limiting conditions for operation are met for the various systems. Conformance to the regulatory criteria for operation with the conditional exemption from the near-EOL MTC measurement is demonstrated and the regulatory limits are not exceeded. Therefore, the margin of safety as defined in the TS [technical specification] is not reduced and 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 standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the request for amendments involves no significant hazards consideration.

Attorney for licensee: A.H. Gutterman, Esq., Morgan, Lewis, & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

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: May 24, 2002.

Description of amendment request: The proposed amendments would allow Mode 2 (startup) operation with two, rather than three, intermediate range monitor channels per trip system.

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

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

The intermediate range monitors (IRMs) monitor neutron flux levels in the reactor core during startup. The IRM detectors are capable of generating a trip signal during a continuous rod withdrawal error in the startup range. However, the IRMs perform no function related to the probability of occurrence of a previously evaluated accident. Also, the IRM trip signal is not necessary to mitigate the limiting control rod withdrawal error. The limiting case assumes the trip signal is generated from the safety-related average power range monitor (APRM). Therefore, the consequences of this previously evaluated abnormal operating transient are not increased.

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

The proposed change reduces the number of required operable IRM channels per trip system from three to two. However, the manner in which the actuation logic functions and the systems respond are unaffected by the proposed change. Furthermore, the IRMs will continue to perform their design function of core monitoring during startup and mitigating nonlimiting transient events postulated to occur during startup. Therefore, the proposed change cannot 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 Bases for Units 1 and 2 Technical Specification Table 3.3.1.1–1 state the ''IRMs are capable of generating trip signals that can be used to prevent fuel damage resulting from abnormal operating transients in the intermediate power (startup) range.'' The proposed change ensures the IRMs will still effectively mitigate these events. The most significant source of reactivity change is due to a control rod withdrawal error. With the proposed change, the IRMs will continue to provide protection against rod withdrawal errors, and peak fuel energy depositions will remain below the 170 cal/gm threshold criterion defined in the Technical Specifications Bases. Therefore, the proposed change does not reduce 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: March 19, 2002, as supplemented on June 3, 2002.

Description of amendment request: The proposed Technical Specification changes involve the removal of the existing scram function and Group 1 isolation valve closure functions of the Main Steam Line Radiation Monitors (MSLRM). An explicit requirement for periodic functional test and calibration of the MSLRM is added to maintain operability of the mechanical vacuum pump (MVP) isolation function. This proposed no significant hazards consideration determination replaces in its entirety the notice published in the **Federal Register** on May 14, 2002 (67 FR 34495).

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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

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

The scram and Group 1 isolation functions of the MSLRMs do not serve as initiators for any of the accidents evaluated in the Updated Final Safety Analysis Report (UFSAR). The MSLRM scram function is not credited in the UFSAR, and the Group 1 isolation trip function of the MSLRMs was only assumed in one design-basis event which was the control rod drop accident. Because these functions are not initiators of accidents, their removal does not increase the probability of occurrence of previously evaluated accidents.

There is no accident analysis that relies on the high radiation scram of the reactor protection system and its removal has no impact on the consequences of accidents previously evaluated. The results of the control rod drop accident analysis remain within approved guidelines, thus any potential increase in consequences would not be considered significant.

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

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

The proposed changes to the plant involve limited changes to protective circuitry, but do not involve any plant hardware changes that could introduce any new failure modes. The changes will not affect non-MSLRM scram and isolation functions. In addition, the MSLRMs will remain active for other trip/isolation functions, and these monitors will still alarm in the control room to alert operators to off-normal conditions.

Therefore, the removal of the Group 1 isolation valve closure and scram functions of the MSLRMs does not create the possibility of a new or different kind of accident than those previously evaluated.

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

The proposed change involves the elimination of the scram and Group I isolation signal from the MSLRMs. Operation under the proposed change will not change any plant operation parameters, nor any protective system setpoints other than removal of these functions. The effects of the control rod drop accident without the MSLRM scram and isolation signal results in doses which remain well within 10 CFR Part 100, ''Reactor Site Criteria,'' limits.

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

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: James W. Clifford.

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

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

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

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

Date of amendment request: June 13, 2002.

Brief description of amendment request: The proposed amendment would revise Technical Specifications Section 4.13.A, ''Inspection Requirements,'' to allow the use of the optimum eddy current probe size when performing steam generator tube inspections. The proposed amendment would also correct several grammatical errors.

*Date of publication of individual notice in **Federal Register:*** June 25, 2002 (67 FR 42806).

Expiration date of individual notice: July 25, 2002.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: April 17, 2001.

Brief description of amendment: The amendment makes editorial and administrative corrections to Technical Specifications (TS) Section 3.3, ''Instrumentation,'' and eliminates minor discrepancies between TS Section 3.3 and other plant licensing basis documents.

Date of issuance: June 25, 2002.

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

Amendment No.: 152.

Facility Operating License No. NPF–62: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 26, 2001 (66 FR 66463). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 25, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 13, 2001.

Brief description of amendments: The amendments revise Item d of TS 5.5.11, ''Ventilation Filter Testing Program (VFTP),'' to lower the maximum allowable differential pressure across the engineered safety features ventilation systems units when tested at the specified system flow rates.

Date of issuance: June 18, 2002.

Effective date: June 18, 2002, and shall be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1–142, Unit 2–142, Unit 3–142.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** February 5, 2002 (67 FR 5325). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 18, 2002.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of application for amendment: November 19, 2001, as supplemented March 27, 2002

Brief description of amendment: The amendment revises Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of the Unit 2 steam generators.

Date of issuance: June 27, 2002.

Effective date: As of the date of issuance to be implemented following the Unit 2 refueling and steam generator replacement outage in spring 2003.

Amendment No.: 230.

Renewed License No. DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** March 19, 2002 (67 FR 12599).

The March 27, 2002, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 27, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 21, 2002.

Brief description of amendment: The amendment authorizes changes to the Updated Final Safety Analysis Report (UFSAR) and the Technical Requirements Manual to eliminate the chlorine detection function from the control center heating, ventilation and air conditioning system. Changes to the UFSAR are subject to the requirements of 10 CFR 50.59; however, the changes were submitted to the Nuclear Regulatory Commission for review and approval since they involve the elimination of an automatic action.

Date of issuance: June 26, 2002.

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

Amendment No.: 147.

Facility Operating License No. NPF–43: Amendment revises the UFSAR and TRM.

Date of initial notice in ***Federal Register:*** April 16, 2002 (67 FR 18643). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 24, 2001.

Brief description of amendment: The amendment deletes License Condition 2.C.(11), which required inspection of the low-pressure turbine discs during the second refueling outage and specified that the frequency of subsequent inspections should be in accordance with the turbine manufacturer's recommendations. License Condition 2.C.(11) is no longer applicable to Fermi 2.

Date of issuance: June 26, 2002.

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

Amendment No.: 148.

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

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64288). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 2002.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–269, 50–270, and 50–287, Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina

Date of application of amendments: June 21, 2000, as supplemented by letters dated April 30 and May 20, 2002.

Brief description of amendments: The amendments authorize changes to the Updated Final Safety Analysis Report Section 10.4.7, ''Emergency Feedwater System.''

Date of Issuance: June 11, 2002.

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

Amendment Nos.: 325/325/326.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments authorized changes to the UFSAR.

Date of initial notice in ***Federal Register:*** July 26, 2000 (65 FR 46008). The supplement dated April 30 and May 20, 2002, provided clarifying information that did not change the scope of the June 21, 2000, application nor the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 11, 2002.

No significant hazards consideration comments received: No.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of application for amendment: April 16, 2001, as supplemented by letters dated November 8, 2001, and February 11, 2002.

Brief description of amendment: The amendment authorizes the licensee to modify the Final Safety Analysis Report (FSAR) to allow an unisolable drain line between the reactor core isolation cooling and the control rod drive/condensate pump rooms and identify the pump room doors and penetration seals that are not watertight. In addition, the change documents the minimum acceptable safe shutdown equipment.

Date of issuance: June 19, 2002.

Effective date: June 19, 2002, and shall be implemented in the next periodic update to the FSAR in accordance with 10 CFR 50.71(e).

Amendment No.: 176.

Facility Operating License No. NPF–21: The amendment revises the FSAR.

Date of initial notice in ***Federal Register:*** May 16, 2001 (66 FR 27175). The November 8, 2001 and February 11, 2002, supplemental letters 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 June 19, 2002.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: April 11, 2002.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed Surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified

Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: June 27, 2002.

Effective date: June 27, 2002.

Amendment No.: 212.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** May 14, 2002 (67 FR 34485). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 27, 2002.

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: March 13, 2002.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: June 10, 2002.

Effective date: As of the date of issuance and shall be implemented in conjunction with the implementation of Amendment No. 215.

Amendment No.: 217.

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

*Date of initial notice in **Federal Register:*** April 30, 2002 (67 FR 21287). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 10, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of application for amendments: August 1, 2001.

Brief description of amendments: These amendments revise Limerick Generating Station's Units 1 and 2 Technical Specifications by deleting Section 6.4, ''Training.''

Date of issuance: June 14, 2002.

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

Amendment Nos.: 160/122.

Facility Operating License Nos. NPF–39 and NPF–85: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 31, 2001 (66 FR 55018). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 14, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 18, 2002.

Brief description of amendment: The proposed amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation, following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: June 26, 2002.

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

Amendment No.: 203.

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

*Date of initial notice in **Federal Register:*** May 14, 2002 (67 FR 34487). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 26, 2002.

No significant hazards consideration comments received: No.

GPU Nuclear Inc., Docket No. 50–320, Three Mile Island Nuclear Station, Unit 2, Dauphin County, Pennsylvania

Date of amendment request: February 8, 2002.

Brief description of amendment request: The amendment would replace referenced control requirements for access to high radiation areas with the actual requirements of 10 CFR Part 20, and would replace the existing Three Mile Island Nuclear Station, Unit 2, Technical Specifications (TS) Section 6.11 with the wording contained in Three Mile Island Nuclear Station, Unit 1, TS Section 6.12.

Date of issuance: June 27, 2002.

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

Amendment No.: 58.

Facility Operating License No. DPR–73: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 2, 2002 (67 FR 15623). The Commission's related evaluation of the amendment is contained in a safety evaluation dated June 27, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 18, 2001, as supplemented by letters dated January 30, and March 1, 2002.

Brief description of amendment: The amendment revises (1) the reference point for reactor vessel level instrumentation specifications to use instrument ''zero'' instead of ''top of active fuel;'' (2) simplifies the safety limits and limiting safety system settings to eliminate specifications that are unnecessary, outdated, or redundant to other Technical Specifications (TSs); (3) changes the reactor coolant system pressure safety limit from 1335 psig to 1332 psig to correct a minor calculation error; and (4) makes corresponding TS Bases changes.

Date of issuance: June 11, 2002.

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

Amendment No.: 128.

Facility Operating License No. DPR–22: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38764). The supplements dated January 30 and March 1, 2002, 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 June 11, 2002.

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: January 10, 2002.

Brief description of amendments: The amendments revise Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a Limiting Condition for Operation following a missed surveillance. The delay period is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: June 19, 2002.

Effective date: June 19, 2002, shall be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 1–153; Unit 2–153.

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

*Date of initial notice in **Federal Register:*** March 5, 2002 (67 FR 10014). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 19, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 22, 2002, as supplemented by letters dated June 10, and June 14, 2002.

Brief description of amendment: This amendment revises Technical Specification (TS) TS 5.5.2.11.f.1.h, ''Steam Generator (SG) Tube Surveillance Program,'' to more clearly delineate the scope of the SG tube inspection required in the tubesheet region. This TS change will apply only to Cycle 12 (Unit 2) and Cycle 11 (Unit 3) operations.

Date of issuance: June 17, 2002.

Effective date: June 17, 2002, to be implemented within 30 days from the date of issuance.

Amendment Nos.: Unit 2—189 ; Unit 3—180.

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

Public comments requested as to proposed no significant hazards consideration: Yes (67 FR 38150 dated May 31, 2002). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by July 1, 2002, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, consultation with the State of California and final determination of no significant hazards consideration are contained in a Safety Evaluation dated June 17, 2002. The June 10, and June 14, 2002, supplemental letters 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.

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: March 25, 2002, as supplemented by the letter dated April 23, 2002.

Brief description of amendments: The proposed change revised the Technical Specification (TS) 3.7.3, ''Feedwater Isolation Valves (FIVs) and Associated Bypass Valves,'' to adopt the NUREG–1431, ''Standard Technical Specifications for Westinghouse Plants,'' Revision 2 version of the specification. The requirements of revised TS 3.7.3 added, among other things, operability and suitable surveillance requirements for Feedwater Control Valves and Associated Bypass Valves and allowed for the extended out-of-service time for one or more FIVs. In addition, a footnote which allowed a one-time extension for Condition A Completion Time, has been deleted because it is no longer applicable.

Date of issuance: June 20, 2002.

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

Amendment Nos.: NPF–87, Amendment No. 97 and NPF–89, Amendment No. 97.

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

*Date of initial notice in **Federal Register:*** May 14, 2002 (67 FR 34492). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 20, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 1st day of July 2002.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 02–16956 Filed 7–8–02; 8:45 am]

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SECURITIES AND EXCHANGE COMMISSION

[Release No. 34–46147; File No. SR–CSE–2002–06]

Self-Regulatory Organizations; Notice of Filing and Immediate Effectiveness of Proposed Rule Change by the Cincinnati Stock Exchange, Inc. Extending a Pilot Revenue Sharing Program for Trading in Nasdaq National Market Securities

June 28, 2002.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (''Act''),[1] and Rule 19b–4 thereunder,[2] notice is hereby given that on June 28, 2002, the Cincinnati Stock Exchange, Inc. (''CSE'' or ''Exchange'') filed with the Securities and Exchange Commission (''Commission'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by CSE. The Commission is publishing this notice to solicit comments on the proposed rule change from interested persons.

I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The Exchange proposes to extend a pilot related to a fee schedule for transactions in Nasdaq National Market securities (''Nasdaq NM Securities'') and to establish a revenue sharing program to reflect recent developments in competitive business strategy. The text of the proposed rule change is below. Additions are in italics, and deletions are in brackets.

Chapter XI

Trading Rules

Rule 11.10 National Securities Trading System Fees

A. Trading Fees (No Change to Text)

* * * * *

(e)*(1)* (No Change to Text)

(2) Tape ''C'' Transactions. Tape ''C'' Transactions are defined as transactions conducted in Nasdaq securities pursuant to unlisted trading privileges (''UTP''). Members will be charged a per share fee for Nasdaq securities based upon the following schedule:

Number of Shares Traded (In a single day)	*Fee Per Share*
0–5 million	*$0.001*
5 million one plus+	*$0.000025*

* * * * *

(l) [Tape ''C'' Transactions. Tape ''C'' Transactions are defined as transactions conducted in Nasdaq securities pursuant to unlisted trading privileges (''UTP''). Members will be charged $.001 per share per side ($1.00/1000 shares), with a maximum charge of $37.50 per firm per side, for Tape C Transactions.]

[Tape ''C'' Transaction Credit. Members will receive a 75 percent pro rata credit on revenue generated by transactions in Tape ''C'' securities.

[(l)]*(m)* (No Change in Text)

[(m)]*(n)* (No Change in Text)

[(n)]*(o)* (No change in Text)

[(o)]*(p)* (No change to text).

[(p)]*(q)* (No change to text)

* * * * *

II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the CSE included statements concerning the purpose of, and the basis for, the proposed rule change and discussed any comments it received on the proposed rule change. The text of these statements may be examined at the places specified in Item IV below. CSE has prepared summaries, set forth in Sections A, B, and C below, of the most significant aspects of such statements.

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.