SUPPLEMENTARY INFORMATION: The meeting will be open to the public up to the seating capacity of the room. The agenda for the meeting is as follows:

—Opening Remarks
—Highlights from Centennial Kick-Off Events
—Web Site Demo
—Aerospace Industries Association (AIA) National Model Rocket Competition
—North Carolina Status
—Next Quarterly Report
—Licensed Products Update
—Aviation Foundation of America's National Air Tours
—Closing Comments

It is imperative that the meeting be held on this date to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitor's register. Due to the increased security at NASA facilities, any members of the public who wish to attend this meeting of the Centennial of Flight Commission must provide their name, date and place of birth, citizenship, social security number, or passport and visa information (number, country of issuance and expiration), business address and phone number, if any. This information is to be provided at least 72 hours (5 p.m. e.d.t. on January 24, 2003) prior to the date of the public meeting. Identification information is to be provided to Beverly Farmarco, (202) 358–1903, *bfarmarc@hq.nasa.gov.* Failure to timely provide such information may result in denial of attendance. Photo identification may be required for entry into the building. Persons with disabilities who require assistance should indicate this in their message. Due to limited availability of seating, members of the public will be admitted on a first-come, first-serve basis. News media wishing to attend the meeting should follow standard accreditation procedures. Members of the press who have questions about these procedures should contact the NASA Headquarters newsroom (202/358–1600).

June W. Edwards,

Advisory Committee Management Officer, National Aeronautics and Space Administration.

[FR Doc. 03–1218 Filed 1–17–03; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL CREDIT UNION ADMINISTRATION

Sunshine Act Meeting

TIME AND DATE: 10 a.m., Thursday, January 23, 2003.

PLACE: Board Room, 7th Floor, Room 7047, 1755 Duke Street, Alexandria, VA 22314–3428.

STATUS: Open.

MATTERS TO BE CONSIDERED:

1. Quarterly Insurance Fund Report.
2. Texas Member Business Loan Rule Proposed Change.

FOR FURTHER INFORMATION CONTACT: Becky Baker, Secretary of the Board, Telephone: 703–518–6304.

Becky Baker,

Secretary of the Board.

[FR Doc. 03–1380 Filed 1–16–03; 2:30 pm]

BILLING CODE 7535–01–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, December 27, 2002 through January 9, 2003. The last biweekly notice was published on January 7, 2003 (68 FR 798).

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 February 20, 2003, 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*.

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

Date of amendment request: November 27, 2002.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TS) and other elements of the licensing bases to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described

[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 paragraphs (d)(1) and (d)(2) regarding petitions to intervene and contentions. For the complete, corrected text of 10 CFR 2.714(d), please see 67 FR 20884; April 29, 2002.

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 TS for nuclear power reactors currently licensed to operate. 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 changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, 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 following NSHC determination in its application dated November 27, 2002.

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

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

The 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 the 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 the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: November 27, 2002.

Description of amendment request: The proposed amendment would delete requirements from the Technical Specifications (TSs) and 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 changes are based on Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a

Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, 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 following NSHC determination in its application dated November 27, 2002.

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

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

The 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 the 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 the margin of safety.

Based upon the reasoning presented above and the previous discussion of the proposed amendment, the NRC staff proposes to determine that the requested amendment does not involve a significant hazards consideration.

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: December 16, 2002.

Description of amendment request: The licensee proposed to amend the Oyster Creek Nuclear Generating Station (OCNGS) Technical Specifications (TSs) regarding the safety limit minimum critical power ratio (SLMCPR) to reflect the results of cycle-specific calculations performed for the current fuel cycle (*i.e.*, Cycle 19), using Nuclear Regulatory Commission (NRC)-approved methodology for determining SLMCPR values. Specifically, the licensee proposed to revise TS 2.1.A, changing the SLMCPR from 1.12 to 1.10 for three-recirculation-loop operation, and from 1.11 to 1.09 for four-or five-recirculation-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. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis is presented below:

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

Before commencement of Cycle 19, the licensee used NRC-approved methods and procedures in Topical Report NEDE–24011–P–A–14, ''General Electric Standard Application for Reactor Fuel'' (GESTAR II) and U.S. Supplement, NEDE–24011–P–A–14–US, dated June 2000, to derive the SLMCPR values for OCNGS, Cycle 19. The revised values were approved by the NRC staff via Amendment No. 233, dated September 26, 2002. Subsequently, the licensee recalculated these SLMCPR values using the methodology in Topical Report NEDC–32694–P–A, ''Power Distribution Uncertainties for Safety Limit MCPR Evaluation,'' and requested to revise these values further by the December 16, 2002, application.

The analysis methodology incorporates cycle-specific parameters. These calculations do not change the operating procedures of OCNGS and have no effect on the probability of an accident initiating event or transient. The basis of the SLMCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and the probability of fuel damage is not increased (*i.e.*, in the event of an accident or transient, the amount of fuel damaged would not be increased as a result of the new SLMCPR values). Furthermore, the proposed new SLMCPR values do not lead to, nor do they arise as a result of, plant design or procedural changes. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The new SLMCPR values for OCNGS Cycle 19 core have been calculated in accordance with the methods and procedures described in NRC-approved topical reports. The proposed new SLMCPR values do not lead to, nor do they arise as a result of, plant design or procedural changes. The changes do not involve any new method for operating the facility and do not involve any facility modifications. As a result, no new initiating events or transients could develop from the proposed changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined in OCNGS's licensing basis will remain the same. The new, cycle-specific SLMCPR values are calculated using NRC-approved methods and procedures that are in accordance with the current fuel design and licensing criteria. The SLMCPR values will remain high enough to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid transition boiling if the limits are not violated, thereby preserving the fuel cladding integrity. Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

Based on the above 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 amendment involves no significant hazards consideration.

Attorney for licensee: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: November 20, 2002.

Description of amendment request: The amendment request reflects organizational changes due to the transfer of the Palisades Plant from Consumers Energy to Nuclear Management Company. The revision reduces redundancy between the Defueled Technical Specifications (DTS) and the Big Rock Point Quality Program Description for Nuclear Power Plants. Other changes are being proposed to correct minor typographical, grammatical, and spelling errors.

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

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

Because this proposed change involves only a change in reporting relationships, and no accidents previously evaluated consider administrative controls, the change does not involve a significant increase in the probability or consequences of an accident previously considered.

2. Will the proposed change create the possibility of a new or different kind of accident [from any other accident] previously evaluated?

The proposed change would result in moving requirements for certain reporting relationships from the Defueled Technical Specifications to the Consumers Energy Quality Program Description for Nuclear Power Plants, Part 1—Big Rock Point Plant (CPC–2A). Because the Topical Report, CPC–2A, requires prior NRC [U.S. Nuclear Regulatory Commission] approval for any changes which would reduce the level of commitment in that document, an equivalent level of NRC oversight is applied to changes to CPC–2A as are applied to changes to Chapter 6 (Administrative Controls) of the Defueled Technical Specifications. Therefore, no changes in administrative controls defined in CPC–2A that might create the possibility of a new or different kind of accident previously evaluated would be permitted by the proposed change.

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

The proposed change stipulates that individuals who perform audits, surveillances and independent safety reviews will report as indicated in CPC–2A, which states that independent safety reviews are performed by the Restoration Safety Review Committee (RSRC). The proposed change involves no significant change in a margin of safety because margins of safety (in the Defueled Technical Specifications) are directly controlled by system design and operation in accordance with Limiting Conditions of Operation, Surveillances and Design Features specified in the Defueled Technical Specifications are affected by this proposed change.

To the extent that design and operation of systems having safety margins might be affected by independent oversight, the following is offered as evidence that no significant reduction in margin of safety will result from the proposed change:

- The Manager, Nuclear Performance Assessment Department (NPAD) and the RSRC both report their findings directly to the Senior Nuclear Officer; therefore there will be no change in the ultimate reporting relationship.
- The membership of NPAD and RSRC consists of individuals who are independent of the plant organization.
- Changes to the Topical Report, CPC–2A that would result in a reduction in level of commitment in the Quality Program Description require a review and approval process equivalent to proposed changes to the administrative controls specified in the Defueled Technical Specifications.
- The requirements for performing onsite and offsite reviews and audits are specified in CPC–2A; the proposed change to the DTS to place the reporting relationship for individuals performing these audits and reviews eliminates redundancy between the Defueled Technical Specifications and CPC–2A.

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., Docket No. 50–423, Millstone Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: December 11, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) related to N–1 loop operation. Specifically, the proposed changes would eliminate N–1 loop operation from particular sections of the TS and would make other changes that are clarifying and/or administrative in nature. In addition, the TS Bases would be revised to address the proposed changes.

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

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

The proposed changes do not alter the way any structure, system, or component functions and would not alter the way [in] which the plant is operated. The proposed changes do not involve any physical plant modifications. The proposed changes incorporate existing plant operational restrictions into the facility Technical Specifications and provide for the removal of information which is not applicable to plant operation.

The proposed allowed outage times (*i.e.* the required action times for Specification 3.4.1.5) are reasonable and consistent with the existing technical specification outage times and consistent with industry guidelines, thereby ensuring affected components are restored in a timely manner. The proposed changes to surveillance requirements are also consistent with existing surveillance frequencies and focus the Technical Specifications on verifying normal plant configurations are maintained. The design basis accidents, including the uncontrolled rod withdrawal from subcritical and boron dilution events, will remain the same postulated events described in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR), and the consequences of these events will not be affected.

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 proposed changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed 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 proposed changes do not introduce any new failure modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce the margin of safety since they have no impact on any accident analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be OPERABLE or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions. The effectiveness of Technical Specifications will be maintained since the changes will not alter the operation of any component or system, nor will the proposed changes affect any safety limits or safety system settings which are credited in a facility accident analysis. Therefore, there is no 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: December 4, 2002.

Description of amendment request: The amendments revise Technical Specification 3.7.6 by changing the minimum combined inventory for Emergency Feedwater from 72,000 gallons to 155,000 gallons and eliminating the condensate storage tank as a source of this inventory.

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:

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures 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:

No. This revision to Technical Specification (TS) 3.7.6 changes the inventory requirements for the Upper Surge Tank (UST) and hotwell. These components provide a suction source to the Emergency Feedwater System (EFW). This increase in inventory from 72,000 gallons to 155,000 gallons increases the required available inventory. This increase in inventory does not affect the probability or consequences of any previously evaluated accident.

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

No. This revision to the combined UST and hotwell inventory increases the required amount of water available to the EFW system. No new or different kind of accident is created by this change as only the required inventory is revised.

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

No. The increase in required UST and hotwell inventory does not reduce the margin of safety. The increase provides the required inventory to ensure that the EFW can provide a Reactor Coolant System cooldown at a rate of 50° F/hour to decay heat removal entry conditions following a reactor trip.

Duke has concluded, based on the above, that there is no significant hazards considerations involved in this amendment request.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: May 1, 2002, as supplemented December 4, 2002.

Description of amendment request: The proposed amendment would extend the applicability of the current Pilgrim Nuclear Power Station (Pilgrim) reactor pressure vessel pressure-temperature (P–T) curves through the end of Operating Cycle (OC) 16. The current P–T curves were approved for use in License Amendment 190, dated April 13, 2001, and are limited to use through the end of OC 14. The proposed change would delete the 20 and 32 Effective Full Power Year (EFPY) curves and replace the wording of the title blocks to allow use through the end of OC 16. The proposed amendment would change Pilgrim Technical Specification Figures 3.6.1, 3.6.2 and 3.6.3.

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

The proposed change involves a request to extend the use of the current reactor pressure vessel P–T curves for two additional OCs. The P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G and Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, and were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor pressure vessel beltline materials. There are no physical changes to the plant or new modes of operation being introduced by the proposed change. Further, the proposed change does not involve a change to any activities or equipment and is not assumed in the safety analysis to initiate any accident sequence. The proposed change does not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the containment of radioactive materials is affected. Additionally, the proposed change will not create any failure mode not bounded by previously evaluated accidents. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The current P–T curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and ASME Code, Section XI, Appendix G, and were approved by the U.S. Nuclear Regulatory Commission for use through OC 14. The proposed change would extend use of the P–T curves for two additional OCs. No new modes of operation are introduced by the proposed change. Plant operation in compliance with the current P–T curves ensures conditions in which brittle fracture of primary coolant pressure boundary materials is avoided. Accidents involving a breach of the primary coolant pressure boundary have previously been evaluated and no other types of accidents associated with the proposed change have been identified. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor pressure vessel beltline materials and are estimated for 48 effective full-power years. The current curves are approved for use through the end of OC 14 (~19 EFPYs) which provides a conservatism factor of 1.7 between the actual EFPYs at the end of OC 14 and the end-of-life curve (32 EFPY). The change would extend the use of the proposed curves to the end of OC 16 (~23 EFPYs) which provides a conservatism factor of approximately 2.0. The actual EFPYs at the end of OC 16 is bounded by the 48 EFPYs estimated for the current curves. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts 02360–5599.

NRC Section Chief: James W. Clifford.

Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc., Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: December 10, 2002

Description of amendment request: The proposed Technical Specification (TS) amendment request changes the diesel fuel specification to a more current revision in TS 4.10.C. The change would also makes administrative revisions to reflect generic position titles in TS 6.0, correct page numbers and titles in the Table on Contents, and delete the General Table of Contents. Bases pages were also revised to reflect the fuel specification revision as well as to make administrative changes to provide clarity and correct a misspelling.

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

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

VY has determined that the probability of occurrence of a previously evaluated accident is not increased because the proposed changes do not impact any accident initiating conditions. The proposed changes will have no significant impact on any safety related structures, systems or components. Additionally, the administrative changes do not affect any system operation or function.

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

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

VY has determined that the proposed changes do not involve any physical alteration of plant equipment and do not change the method by which any safety-related system performs its function. No new or different types of equipment will be installed. The proposed changes do not create any new accident initiators or involve an activity that could be an initiator of an accident of a different type.

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

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

VY has determined that the proposed changes do not alter the basic operation of process variables, systems, or components as described in the safety analysis. No new equipment is introduced.

The proposed changes do not impact design margins of any system to perform its intended safety functions. There is no physical or operational change being made which would alter the sequence of events, plant response, or margins in existing safety analyses. The proposed changes result in no impact on analyzed accident event precursors or effects. These proposed changes do not alter the physical design of the plant. There is no change in methods of operation.

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

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

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

NRC Section Chief: James W. Clifford.

Exelon Generation Company, LLC, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois; Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois; Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2 Docket Nos. 50–277 and 50–278, Peach Bottom Atomic Power Station, Units 2 and 3, York County, Pennsylvania; Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: November 27, 2002.

Description of amendment request: The proposed amendments delete requirements from the technical specifications (TS) and 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 TS for nuclear power reactors currently licensed to operate. 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 changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The NRC staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, 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 following NSHC determination in its application dated November 27, 2002.

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

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

The 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 the 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 the margin of safety.

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

NRC Section Chiefs: Anthony J. Mendiola, James W. Clifford.

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

Date of amendment request: October 30, 2002.

Description of amendment request: The proposed amendment deletes requirements from the technical specifications (TS) and other elements of the licensing bases to maintain a Post Accident Sampling System (PASS).

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

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

The 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 Three Mile Island (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 23 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, the Emergency Operating Procedures (at PNPP, these procedures are titled the Plant Emergency Instructions), and site survey monitoring that support modification of Emergency Plan Protective Action Recommendations (PARs). Therefore, the elimination of PASS requirements from Technical Specifications does not involve a significant increase in the consequences of any accident previously evaluated.

2. The proposed change would 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.

3. The proposed change will not involve a significant reduction in the 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 do not rely on PASS are designed to provide rapid assessment of current reactor core conditions and the trending 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 the 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.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: March 22, 2002, as supplemented May 13, June 24, July 29, and December 20, 2002.

Description of amendment request: The proposed amendment would revise Technical Specifications (TSs) Surveillance Requirement (SR) 4.0.3 to extend the delay period, before entering a Limiting Condition for Operation (LCO), following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 4.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process (CLIIP). The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated March 22, 2002, as supplemented May 13, June 24, July 29, and December 20, 2002.

The proposed amendment would also add a requirement for a TS Bases Control Program to the administrative controls section of TSs. This change is necessary to be consistent with the CLIIP and is also consistent with the TS Bases Control Program presented in Section 5.5 of NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plants.'' The licensee provided its analysis of the issue of NSHC for this proposed change in its application.

The proposed amendment would also modify SR 4.0.1, and its associated Bases, to link it with SR 4.0.3. The modification to SR 4.0.1 is consistent with NUREG–1431, Revision 2, ''Standard Technical Specifications Westinghouse Plants.'' The licensee provided its analysis of the issue of NSHC for this proposed change in its application.

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

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

[CLIIP Change]

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

[Addition of TS Bases Control Program and Changes to SR 4.0.1] The proposed changes to adopt the ITS [Improved Standard Technical Specifications] wording for Specification 4.0.1 and formally adopt a [TS] Bases Control Program are administrative in nature and do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, configuration of the facility or the manner in which it is operated. The proposed changes do not alter or prevent the ability or structures, systems, or components to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Seabrook Station Updated Final Safety Analysis Report (UFSAR).

Future changes to the TS Bases will continue to be administratively controlled pursuant to the provisions of 10 CFR 50.59. The TS Bases is a licensee-controlled document that contains bases information for the [TS]. Future changes to the information contained in the TS Bases will be reviewed and approved in accordance with the FPLE Seabrook Regulatory Compliance Manual and TS Section 6.7.6j (TS Bases Control Program) of the Seabrook Station [TS]. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

[CLIIP Change]

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

[Addition of TS Bases Control Program and Changes to SR 4.0.1]

The proposed changes do not alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. There are no changes to the source term or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. The proposed changes have no adverse impact on component or system interactions. The proposed changes will not adversely degrade the ability of systems, structures and components important to safety to perform their safety function nor change the response of any system, structure or component important to safety as described in the UFSAR. The proposed changes are administrative in nature and do not change the level of programmatic and procedural details of assuring operation of the facility in a safe manner. Since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

[CLIIP Change]

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function.

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.

[Addition of TS Bases Control Program and Changes to SR 4.0.1]

There is no adverse impact on equipment design or operation and there are no changes being made to the [TS] required safety limits or safety system settings that would adversely affect plant safety. The proposed changes are administrative in nature and do not reduce the level of programmatic or procedural controls associated with the activities presently performed via the aforementioned surveillance requirements.

Future changes to the TS Bases information will be reviewed and approved in accordance with Seabrook Station [TS], Section 6.7, and as outlined in [FPLE Seabrook's] Regulatory Compliance programs. Specifically, changes to the Seabrook Station [TS] Bases require an evaluation pursuant to the provisions of 10 CFR 50.59 and review and approval by the Station Operation Review Committee (SORC) prior to implementation.

Therefore, formal adoption of a TS-required TS Bases Control Program and adoption of ITS wording for Specification 4.0.1 do not involve a significant reduction in the margin of safety provided in the existing specifications.

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 NSHC.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 15, 2002.

Description of amendment request: The proposed amendment would revise the Donald C. Cook Nuclear Plant, Unit 2, operating license and Technical Specifications to increase the licensed power level to 3468 Mega Watts Thermal (MWt), or 1.66 percent greater than the current level of 3411 MWt.

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated—In support of this Measurement Uncertainty Recapture (MUR) power uprate, a comprehensive evaluation was performed for nuclear steam supply system (NSSS) and balance of plant systems and components and analyses that could be affected by this change. A power calorimetric uncertainty calculation was performed, and the effect of increasing plant power by 1.66 percent on the plant's design and licensing basis was evaluated. The result of these evaluations is that all plant components will continue to be capable of performing their design function at an uprated core power of 3468 MWt. In addition, an evaluation of the accident analyses demonstrates that applicable analysis acceptance criteria continue to be met. No accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change.

Consequences of an Accident Previously Evaluated—This change does not affect the release paths, the frequency of release, or the source term for release for any accidents previously evaluated in the Updated Final Safety Analysis Report. Structures, systems, and components (SSC) required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3468 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 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 single failures are introduced as a result of the proposed changes. The installation of the Caldon Leading Edge Flow Meter CheckPlus™ system has been analyzed, and failures of this system will have no adverse effect on any safety-related system or any SSCs required for transient mitigation. SSCs 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.

This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 3468 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the feedwater flow input to the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at a core power of 3468 MWt. Credible malfunctions continue to be bounded by the current accident analysis of record or evaluations that demonstrate that applicable acceptance criteria continue to be met.

Therefore, the proposed changes do 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 margins of safety associated with this MUR Uprate Program are those pertaining to core power. This includes those associated with the fuel cladding, Reactor Coolant System pressure boundary, and containment barriers. A comprehensive engineering review was performed to evaluate the 1.66 percent increase in the licensed core power from 3411 MWt to 3468 MWt. The 1.66 percent increase required that revised NSSS design thermal and hydraulic parameters be established, which then served as the basis for all of the NSSS analyses and evaluations. This engineering review concluded that no design transient modifications are required to accommodate the revised NSSS design conditions. NSSS systems and components were evaluated and it was concluded that the NSSS equipment has sufficient margin to accommodate the 1.66 percent power uprate. NSSS accident analyses were evaluated for the 1.66 percent power uprate. In all cases, the evaluations demonstrate that the applicable analyses acceptance criteria continue to be met. As such, the margins of safety continue to be bounded by the current analyses of record for this change.

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

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

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

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

NRC Section Chief: L. Raghavan.

Nine Mile Point Nuclear Station, LLC (NMPNS), Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of amendment request: December 19, 2002.

Description of amendment request: The proposed amendment would update and clarify the Technical Specifications (TSs) requirements for demonstrating shutdown margin (SDM). The proposed changes incorporate new, more restrictive, SDM limits; add the required limiting condition for operation (LCO) actions if the SDM is not met; and also add the surveillance requirements for verifying the SDM. These LCO actions and surveillance requirements are not currently specified in the TSs. The revised SDM limits account for the uncertainty in the demonstration of adequate SDM analytically or by measurement. The proposed changes also eliminate the unnecessary restriction requiring SDM demonstration in the cold shutdown condition. The option for SDM demonstration in the cold shutdown condition is retained consistent with the existing special test exception.

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.

Adequate SDM provides assurance that inadvertent criticalit[y] and potential control rod drop accidents (CRDAs) involving high worth control rods will not cause significant fuel damage. The SDM is not an accident initiator and, as such, will have no effect on the probability of an accident. The proposed changes incorporate more restrictive SDM limits and provide the necessary actions and verifications to assure that there will be no adverse effect on the initial conditions and assumptions of the accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR). The proposed changes do not involve physical changes to the plant or introduce any new modes of operation. Accordingly, continued assurance is provided that the process variables, structures, systems, and components are maintained such that there will be no degradation of any fission product barrier which could increase the radiological consequences of an accident. Therefore, the proposed changes do 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 changes to the SDM limits and requirements will have no adverse effect on the design or assumed accident performance of any structure, system, or component, or introduce any new modes of system operation or failure modes. Moreover, the proposed changes will have no impact on conformance to 10 CFR [Code of Federal Regulations] 50, Appendix A, General Design Criterion 26 (GDC 26), in that the control rods will continue to satisfy the SDM requirements and provide assurance that the reactor can be made subcritical from all applicable operating conditions, transients, and design basis events. 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 change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes provide separate SDM limits for testing consistent with the Improved Standard Technical Specifications (NUREG–1433 and NUREG–1434) where the highest worth control rod is determined analytically (0.38% Dk/k) or by measurement (0.28% Dk/k). The proposed SDM limits are more restrictive than the current limit (0.25% Dk/k) and account for the uncertainty in the demonstration of SDM by testing. The SDM will continue to account for changes in core reactivity during the fuel cycle. Therefore, the margin of safety is increased relative to the SDM assumptions for the control rod withdrawal error transient and CRDA analyses.

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

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

Attorney for licensee: 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 No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: December 19, 2002.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) reporting requirements for the discovery of defective or degraded steam generator tubes.

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

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

Response: No.

The proposed changes do not have any effect on structures, systems, and components (SSCs) of the Kewaunee Nuclear Power Plant. The changes do not affect plant operations, any design function or an analysis that verifies the capability of an SSC to perform a design function. The changes do not change any previously evaluated accidents in the updated safety analysis report (UFSAR). As these changes are administrative, there is no increase in the probability and consequences of analyzed accidents.

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

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

Response: No.

The proposed changes are administrative and do not change the design function or operation of any plant SSCs. The proposed changes do not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

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

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

Response: No.

The proposed changes modify NRC reporting requirements only. The changes do not exceed or alter a design basis or safety limit or significantly reduce the margin of safety.

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

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., Esq., Shaw Pittman, Potts & Trowbridge, 2300 N. Street, NW, Washington, DC 20037–1128.

NRC Section Chief: L. Raghavan.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia

Date of amendment request: December 4, 2002.

Description of amendment request: The proposed amendment changes the Hatch Unit 2 turbine building high temperature primary containment isolation value specified in Technical Specification Table 3.3.6.1–1, Item 1f.

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[?]

This TS [Technical Specification] revision request changes the allowable value for the turbine building high temperature primary containment isolation. The setpoint at which the isolation occurs has nothing to do with preventing a system break; therefore, this proposed change will not change the probability of occurrence of a small primary coolant system break.

For the turbine building high temperature primary containment isolation, the analytical limit has been calculated at 207°F with the allowable value at 200°F. The calculation supporting these values accounts for instrument uncertainties thus confirming that adequate margin exists between the allowable value and the analytical limit. Accordingly, the consequences of a small primary system break are not significantly increased.

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

Changing an allowable value does not introduce any new operating modes for any plant system or piece of equipment. All plant systems will continue to be operated, tested and maintained as before, and within their licensing and design basis. As a result, no new failure modes are introduced and the possibility of a new or different type of accident is not created.

3. [Does] the [* * *] proposed [* * *] change involve a significant decrease in the margin of safety[?]

Increasing the allowable value by 6°F does not result in a significant reduction in a margin of safety. A formal calculation was performed which justified an analytical limit of 207°F. This calculation determined the analytical limit based on a primary leak into the turbine building and confirmed that the allowable value adequately protects the analytical limit. As a result, the margin of safety is not significantly reduced.

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.

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*.

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

Date of application for amendment: February 14, 2002, as supplemented on September 9, 2002.

Brief description of amendment: The amendments revised the Millstone Power Station, Unit No. 2 (MP2) and 3 (MP3) Technical Specifications (TSs) by relocating selected MP2 and MP3 TSs related to the Reactor Coolant System and Plant Systems to the respective unit's Technical Requirements Manual.

The amendment does not address changes to MP2 TS 3/4.7.10, ''Snubbers,'' and MP3 TSs 3/4.7.10, ''Snubbers,'' and 3/4.7.14, ''Area Temperature Monitoring,'' as described by the application dated February 14, 2002, because these proposed TSs changes were withdrawn by the supplement dated September 9, 2002.

Date of issuance: January 2, 2003.

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

Amendment Nos.: 272 and 214.

Facility Operating License Nos. DPR–65 and NPF–49: This amendment revised the TSs.

Date of initial notice in ***Federal Register:*** April 16, 2002 (67 FR18645). The September 9, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 2, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: September 12, 2002, as supplemented by letter dated December 30, 2002.

Brief description of amendments: The amendments temporarily revised Technical Specifications (TS) 3.5.2, ''Emergency Core Cooling System;'' TS 3.6.6, ''Containment Spray System;'' TS 3.7.5, ''Auxiliary Feedwater System;'' TS 3.7.7, ''Component Cooling Water System;'' TS 3.7.8, ''Nuclear Service Water System;'' and TS 3.8.1, ''AC Sources.''

Date of issuance: January 7, 2003.

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

Amendment Nos.: 203 & 196.

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

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63692). The supplement dated December 30, 2002, provided clarifying information that did not change the scope of the September 12, 2002, application, nor the initial no significant hazard consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 7, 2003.

No significant hazards consideration comments received: No

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

Date of application for amendments: August 29, 2002. *Brief description of amendments:* The amendments revise Technical Specification (TS) 3.8.4.7, to modify the note to eliminate the ''once per 60 months'' restriction on replacing the battery service test by the battery modified performance discharge test.

Date of issuance: January 9, 2003.

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

Amendment Nos.: 204 & 197.

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

Date of initial notice in ***Federal Register:*** November 12, 2002 (67 FR 68733). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 9, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 22, 2002.

Brief description of amendment: The amendment deletes a reference to Section 2.E in Section 2.F of Facility Operating License No. NPF–21. Section 2.E requires the licensee to fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans. Section 2.E is redundant because the reporting requirements and criteria for the physical security programs are specified in 10 CFR 73.71 and Appendix G of 10 CFR Part 73.

Date of issuance: January 9, 2003.

Effective date: January 9, 2003 to be implemented within 60 days from the date of issuance.

Amendment No.: 183.

Facility Operating License No. NPF–21: The amendment revised the operating license.

Date of initial notice in ***Federal Register:*** December 10, 2002 (67 FR 75871). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 9, 2003.

No significant hazards consideration comments received: No.

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: August 21, 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: January 2, 2003.

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

Amendment No.: 127.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 1, 2002 (67 FR 61679). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 2, 2003.

No significant hazards consideration comments received: No.

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

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

Brief description of amendment: The amendment provides additional information to support a modification to Technical Specification 3.4.7 and limits Reactor Coolant System activity permitted by the ACTION statement to 60 microcuries per gram at all power levels. The letdown line break accident analysis in the Final Safety Analysis Report was also changed.

Date of issuance: January 8, 2003.

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

Amendment No.: 184.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications and Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** October 28, 2002 (67 FR 66009). The August 27, 2002, supplemental letter provided additional information and revised the no significant hazards consideration determination. The original **Federal Register** notice was published on November 28, 2001 (66 FR 56504), but was superceded by the October 28, 2002 publication.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 8, 2003.

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: June 26, 2002, as supplemented September 12, 2002.

Brief description of amendments: Extend the use of the pressure-temperature limits in Technical Specification Figure 3.4.6.1–1 to 32 effective full power years.

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

Effective date: January 2, 2003.

Amendment Nos. 163 and 125.

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

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50953). The supplement dated September 12, 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 amendments is contained in a Safety Evaluation dated January 2, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–171, Peach Bottom Atomic Power Station, Unit 1, York County, Pennsylvania

Date of application for amendment: May 21, 2002.

Brief description of amendment: This proposed amendment will revise the Peach Bottom Atomic Power Station, Unit 1, License and Technical Specifications (TS) to: (1) Delete License Condition C(4) to reflect satisfaction of the minimum decommissioning trust fund amount at the time of transfer of the Facility Operating License; (2) revise License Condition C(5)(d) to reflect 30 days prior written notification to the Director of Nuclear Material Safety and Safeguards before modification of the decommissioning trust agreement in any material respect; (3) delete TS 2.1(B)3 and TS 2.4(b) to eliminate inconsistencies with reporting requirements in 10 CFR 20.2202, 50.73, and 73.71; (4) revise TS 2.2 to refer to the Facility Operating License; and (5) revise TS 2.3 to refer to the radiological hazards associated with the facility.

Date of Issuance: December 26, 2002.

Effective Date: On the date of issuance of this amendment and must be fully implemented no later than 30 days from the date of issuance.

Amendment No.: 11.

Facility Operating License No. DPR–12: Amendment revised the License and TS with respect to administrative procedures or requirements.

Date of initial notice in ***Federal Register:*** October 1, 2002 (67 FR 61682). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated December 26, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 1, 2001, as supplemented by letters dated June 13, 2001, May 20, 2002, and June 28, 2002.

Brief description of amendments: These amendments revised TS 3.7.1, ''Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS),'' to add operability requirements and surveillance requirements for the UHS spray bypass and large array valves, and reduce the allowed Completion Times for the conditions applicable to the RHRSW system.

Date of issuance: December 30, 2002.

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

Amendment Nos.: 206 and 180.

Facility Operating License Nos. NPF–14 and NPF–22: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 5, 2001 (66 FR 46481). The June 13, 2001, May 20, 2002, and June 28, 2002, letters provided additional information that clarified the application, but 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 amendments is contained in a Safety Evaluation dated December 30, 2002.

No significant hazards consideration comments received: No.

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: December 3, 2001, as supplemented by letter dated August 29, 2002.

Brief description of amendments: The amendments revise TS 3.7.1.2, ''Auxiliary Feedwater System,'' to better reflect the four train auxiliary feedwater (AFW) system design at STP. Specifically, the changes specify the same allowed outage time (AOT) for any one inoperable motor-driven pump, regardless of train. The amendments also extend the AOT for one inoperable motor-driven pump from 72 hours to 28 days. A sentence has also been added to Action d. stating that Limiting Condition for Operation (LCO) 3.0.3 and all other LCO actions requiring Mode changes are suspended until one of the four inoperable AFW pumps is restored to operable status. There is also an administrative change in the wording of the LCO to clarify that there are only four AFW pumps in each STP unit.

Date of issuance: December 31, 2002.

Effective date: December 31, 2002.

Amendment Nos.: Unit 1—146; Unit 2—134.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2930). The supplement provided additional information that clarified the application, did not expand the scope as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 31, 2002.

No significant hazards consideration comments received: No.

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: April 8, 2002.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.4.16, ''RCS [Reactor Coolant System] Specific Activity,'' to lower the Limiting Condition For Operation and associated Surveillance Requirements for Dose Equivalent Iodine-131 in the RCS from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm.

Date of issuance: January 6, 2003.

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

Amendment Nos.: 102 and 102.

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

Date of initial notice in ***Federal Register:*** June 11, 2002 (67 FR 40026). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 6, 2003.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50–338, North Anna Power Station, Unit 1, Louisa County, Virginia

Date of application for amendment: December 7, 2001, as supplemented by letters dated June 28 and July 25, 2002.

Brief description of amendment: This amendment permits a one-time extension of the current 10-year Title 10 of the Code of Federal Regulations Part 50, Appendix J, Option B, Type A test interval from April 3, 2003, to April 2, 2008.

Date of issuance: December 31, 2002.

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

Amendment No.: 234.

Facility Operating License No. NPF–4: Amendment changes the Technical Specifications.

Date of initial notice in ***Federal Register:*** April 30, 2002 (67 FR 21295). The supplemental letters dated June 28 and July 25, 2002, contained clarifying information only and did not change the proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 31, 2002.

No significant hazards consideration comments received: No.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 13th day of January 2003.

John A. Zwolinski,

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

[FR Doc. 03–1161 Filed 1–17–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on February 5, 2003, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, February 5, 2003—1 p.m. until the conclusion of business

The Subcommittee will discuss proposed ACRS activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301/415–7364) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: January 13, 2003.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 03–1221 Filed 1–17–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Materials and Metallurgy; Notice of Meeting

The ACRS Subcommittees on Materials and Metallurgy will hold a meeting on February 5, 2003, Room T–