NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting Notice

DATE: Weeks of August 5, 12, 19, 26, September 2, 9, 2002.

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

STATUS: Public and Closed.

MATTER TO BE CONSIDERED:

Week of August 5, 2002

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

Week of August 12, 2002—Tentative

Tuesday, August 13, 2002

9:25 a.m. Affirmation Session (Public Meeting) (if needed).

9:30 a.m. Briefing on Special Review Group Response to the Differing Professional Opinion/Differing Professional View (DPO/DPV) Review (Public Meeting) (Contact: John Craig, 301–415–1703.

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

Week of August 19, 2002—Tentative

Wednesday, August 21, 2002

9:30 a.m. Briefing on NRC International Activities (Public Meeting) (Contact: Janice Dunn Lee, 301–415–1780).

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

1:55 p.m. Affirmation Session (Public Meeting) (if needed).

2 p.m. Meeting with Organization of Agreement States (OAS) and Conference of Radiation Control Program Directors (CRCPD) (Public Meeting) (Contact: John Zabko, 301–415–2308).

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

Week of August 26, 2002—Tentative

There are no meetings scheduled for the Week of August 26, 2002.

Week of September 2, 2002—Tentative

There are no meetings scheduled for the Week of September 2, 2002.

Week of September 9, 2002—Tentative

There are no meetings scheduled for the Week of September 9, 2002.

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

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ADDITIONAL INFORMATION: By vote of 4–0 on July 31, the Commission determined pursuant to U.S.C. 552b(e) and § 9,107(a) of the Commission's rules that ''Affirmation of Pacific Gas & Electric Co. (Diablo Canyon Power Plant, Units, 1 and 2); Multiple Petitions to intervene'' be held on August 1, and on less than one week's notice to the public.

* * * * *

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

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

Dated: August 1, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–19913 Filed 8–2–02; 11:13 am]

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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, July 12, 2002, through July 25, 2002. The last biweekly notice was published on July 23, 2002 (67 FR 48213).

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 September 5, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714,[1] which is available at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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. 50–219, Oyster Creek Nuclear Generating Station, Ocean County, New York

Date of amendment request: June 26, 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 next 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 1.09 to 1.12 for three-recirculation-loop operation, and to 1.11 for four-or five-recirculation-loop operation. The proposed amendment would also editorially revise references to topical reports which document the approved methodology, and make editorial corrections to the TSs.

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

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 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. The balance of the changes is purely administrative. 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 balance of the changes is purely administrative. 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.

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

Date of amendments request: June 26, 2002.

Description of amendments request: The proposed amendment would revise the Technical Specifications (TS) to revise the reactor coolant system pressure-temperature limit curves for operation to 32 effective full-power years (EFPY).

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 license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. Development of the revised BSEP, Unit 1 and 2 pressure-temperature limits was performed using the approved fracture toughness methodologies of 10 CFR 50, Appendix G; the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Appendix G; and ASME Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1.'' The revised pressure-temperature limits were also developed using NRC Regulatory Guide 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' March 2001, for evaluating neutron fluence and NRC Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials,'' for evaluating predicted irradiation effects on vessel beltline materials. Use of these methods provides compliance with the intent of 10 CFR 50, Appendix G, and provides adequate protection against nonductile-type fractures of the reactor pressure vessel. Therefore, the probability of occurrence of a previously analyzed event is not significantly increased.

The consequences of a previously evaluated accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the accident, the availability and successful functioning of the equipment assumed to operate in response to the accident, and the setpoints at which these actions are initiated. The proposed revisions do not impact the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated, and there are no adverse effects on the factors contributing to offsite and onsite dose. The proposed changes to the pressure-temperature limits curves do not affect the performance of any equipment used to mitigate the consequences of a previously evaluated accident. Also, the proposed changes do not affect setpoints that initiate protective or mitigative actions. Based on the above, the proposed changes to the pressure-temperature limits curves do not significantly increase the consequences of a previously evaluated accident.

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

The changes extend the pressure-temperature limits for use up to 32 EFPY of operation while providing adequate protection against a nonductile-type fracture of the reactor pressure vessel. Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This proposed license amendment does not involve any facility modifications, and plant equipment will not be operated in a different manner. Also, no new initiating events or transients result from the pressure-temperature limits curves changes. As a result, no new failure modes are being introduced. Therefore, the proposed changes to the pressure-temperature limits curves will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through the design of the plant structures, systems, and components; through the parameters within which the plant is operated; through the establishment of setpoints for actuation of equipment relied upon to respond to an event; and through margins contained within the safety analyses. The proposed changes to the pressure-temperature limit curves do not adversely impact the performance of plant structures, systems, components, and setpoints relied upon to respond to mitigate an accident. The revised pressure-temperature limits were developed using the approved fracture toughness methodologies of 10 CFR 50, Appendix G; the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Appendix G; and ASME Code Case N–640, ''Alternative Reference Fracture Toughness for Development of P–T Limit Curves for ASME Section XI, Division 1.'' The proposed changes are acceptable because the ASME guidance maintains the relative margin of safety commensurate with that which existed at the time that the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, was approved in 1974. In addition, the revised pressure-temperature limits were also developed using NRC Regulatory Guide 1.190, ''Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,'' March 2001, for evaluating neutron fluence and NRC Regulatory Guide 1.99, Revision 2, ''Radiation Embrittlement of Reactor Vessel Materials'' for evaluating predicted irradiation effects on vessel beltline materials. Use of these methods has provided revised pressure-temperature limit curves that will ensure that the reactor pressure vessel materials continue to behave in a non-brittle manner, thereby preserving the original safety design bases[.] No plant safety limits, setpoints, or design parameters are adversely affected by the proposed changes to the pressure-temperature limit curves. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

Attorney for licensee: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Kahtan Jabbour, Acting.

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

Date of amendments request: July 2, 2002.

Description of amendments request: The proposed amendments would revise the Technical Specifications (TS) to change the administrative controls of TS 5.7, ''High Radiation Area.'' The proposed changes would be consistent with the guidance of Regulatory Guide 8.38, ''Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,'' Section C, Regulatory Position 2.4, Alternative Methods for Access Control, with the exception that ''should'' would be changed to ''shall.''

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

The changes are administrative and affect personnel access control requirements for high radiation areas. The changes do not affect the operation, physical configuration, or function of plant equipment or systems. The changes do not impact the initiators or assumptions of analyzed events; nor do they impact the mitigation of accidents or transient events. Therefore, these changes do not increase the probability or consequences of an accident previously evaluated.

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

The changes are administrative and affect personnel access control requirements for high radiation areas. The changes do not alter plant configuration, require installation of new equipment, alter assumptions about previously analyzed accidents, or impact the operation or function of plant equipment or systems. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The changes are administrative and affect personnel access control requirements for high radiation areas. The changes do not impact any safety assumptions; nor do the changes have the potential to reduce any margin of safety as described in the BSEP TS Bases. The proposed changes maintain an equivalent level of protection for radiation workers and, thereby, provide reasonable assurance that individuals will not exceed regulatory dose limits. The proposed changes are consistent with: (1) the guidance of Regulatory Guide (RG) 8.38, ''Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,'' Section C, Regulatory Position 2.4, Alternative Methods for Access Control, with the exception that ''should'' has been changed to ''shall''; (2) the BSEP TSs prior to conversion to Improved Standard Technical Specifications; and (3) other nuclear plants' existing TSs, including the Crystal River, H. B. Robinson, and Shearon Harris nuclear plants.

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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Kahtan Jabbour, Acting.

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

Date of amendment request: July 8, 2002.

Description of amendment request: The amendment would revise Technical Specification (TS) 3/4.8.1.1, ''Electrical Power Systems—A.C. Sources—Operating'' and TS 3/4.8.1.2, ''Electrical Power Systems—A.C. Sources—Shutdown'' by revising the minimum level to a volume-based indication versus a level-based indication.

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

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

The Harris Nuclear Plant (HNP) Technical Specification (TS) Bases for Electrical Power Systems—A. C. Systems states that; ''A separate day tank containing a minimum of 1457 gallons of fuel, which is equivalent to a minimum indicated level of 40% * * *'' and, the asterisked note states; * * * Minimum indicated level with a fuel oil specific gravity of 0.83 and the level instrumentation calibrated to a reference specific gravity of 0.876.'' These changes do not modify the design or operation of Structures, Systems, and Components (SSCs) that could initiate an accident. The minimum volume of fuel in the day tank is unchanged by this amendment and consequently would not impact the probability or consequences of any accident scenario.

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

The proposed changes do not involve new plant components or procedures, but only revise existing Technical Specification Limiting Condition for Operation Requirements. No significant impact on any postulated accident is made due to this change since the required fuel oil volume is not changed and the level indication for the operations personnel is not changed. These changes do not modify the design or operation of Structures, Systems, and Components (SSCs) that could initiate an accident.

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

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

The proposed changes do not affect the design or operation of safety related components relied upon to automatically mitigate the consequences of a design basis event. The day tank level specified in TS is not accurate for all fuel oil specific gravities so these changes provide better monitoring capability by reducing the possibility of confusion. Indicated day tank level is used to determine volume by comparing the indicated level to the day tank curve using the actual specific gravity of the fuel. The Diesel Generator day tank minimum volume is not altered by these changes and therefore there * * * is no significant impact on any safety system and these changes do not reduce the margin of safety.

Based on these considerations, 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Kahtan N. Jabbour, Acting.

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: July 11, 2002.

Description of amendment request: The proposed amendments would revise the Technical Specifications to make several administrative 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:

First Standard

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

No. This license amendment request makes editorial corrections to several Oconee Technical Specifications. These corrections are solely administrative in nature. The deletion of the Reactor Building Engineered Safeguards Channels, as proposed in the change to the Technical Specification 3.3.6, Engineered Safeguards Protective System Manual Initiation, was investigated through Duke's corrective action program and also confirmed to be administrative in nature. Therefore, all the changes contained in this license amendment request are administrative in nature and have no impact on any accident probabilities or consequences.

Second Standard

Would implementation of this amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

No. There are no new accident causal mechanisms created as a result of the implementation of this license amendment request. No changes are being made to the plant which will introduce any new accident causal mechanisms. This amendment request only makes administrative changes and does not impact any plant systems that are accident initiators; therefore, no new accident types are being created.

Third Standard

Would implementation of this statement involve a significant reduction in a margin of safety?

No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. The changes proposed in this license amendment request are administrative in nature and do not affect the performance of the barriers. Consequently, no safety margins will be impacted.

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

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

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: January 10, 2002.

Description of amendment request: Energy Northwest is requesting changes to the technical specifications (TS) to reflect the application of a 24-month surveillance test interval (STI) to coincide with its intention to implement a 24-month fuel cycle.

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 extension of the intervals to 24 months for the subject SRs [surveillance requirements] does not impact the ability of any of the equipment to function as assumed in the Columbia Generating Station accident analysis. None of the equipment within the scope of analysis for this TS amendment request performs a function in any of the systems required for safe shutdown as described in section 7.4 of the Columbia Generating Station FSAR [Final Safety Analysis Report]. Historical maintenance and surveillance data as well as projected instrument drift indicate the proposed amendment will not affect performance or reliability of the equipment tested to meet the requirements of these SRs. Therefore, the extension of the surveillance intervals does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

An event related to surveillance testing Frequency or instruments drifting beyond Allowable Values is not postulated in the Columbia Generating Station accident analysis. None of the analyses performed for this amendment request indicate an increase in the probability of equipment failure resulting from the surveillance interval extension. Because all of the equipment related to the proposed SR interval extensions is expected to function normally during the longer intervals, extending the subject SRs does not introduce any new accident initiators.

Therefore, the operation of Columbia Generating 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.

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

The proposed amendment to the Technical Specifications will extend the intervals at which testing is performed to meet the requirements of the selected SRs. The overall effect of the extensions on safety is small due to other more frequent testing that is performed on the same equipment, projected instrument drift that is bounded by the current setpoint analysis, or the existence of redundant mechanical or electrical components. Reviews of historical surveillance and maintenance records indicate there is no evidence of time-related failures. The proposed amendment does not impact the performance of any system, structure, or component relied upon for accident mitigation. The proposed surveillance interval extensions do not impact any safety analysis assumptions or results.

Therefore, operation of Columbia Generating Station in accordance with the proposed amendment will 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: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: June 24, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) surveillance requirements (SR) 3.7.7.1 and SR 3.7.7.2. Specifically, SR 3.7.7.1 would be changed to require the verification of the city water tank volume rather than city water header pressure and increase the SR frequency from 12 hours to 24 hours. SR 3.7.7.2 would be revised to require all city water header isolation valves are open rather than only the one header supply isolation valve.

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

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

Response: No.

The current TS surveillance to verify City Water (CW) header pressure did not provide assurance that adequate volume of water was available in the City Water Tank (CWT) as an alternate source of cooling if Condensate Storage Tank (CST) was not available. The CST is not designed to withstand the effect of a tornado-generated missile. However, the Auxiliary Feedwater System (AFS) is provided sufficient redundancy of water supplies such that an alternate source of water from the CWT is available in the event the CST is damaged by a tornado-generated missile. The proposed amendment to verify CWT volume is ≥360,000 gallons would ensure that adequate volume of CW is available in the CWT to cool the RCS [reactor coolant system] from 102% rated thermal power to RHR [residual heat removal] entry conditions in 10 hours, if the CST is unavailable or depleted for any reason. The surveillance frequency for the CWT volume is 24 hours. The proposed amendment to change SR 3.7.7.2 to include additional isolation valves that are in the flow path from CWT to AFS suction would ensure that all applicable isolation valves in the flow path are properly positioned. Thus, the proposed amendment involves changes to the Technical Specifications that would properly reflect the Surveillance Requirements for CWT. The CWT is not an initiator of any accident addressed in the FSAR [Final Safety Analysis Report] and the proposed amendment does not have any change to the accident analysis addressed in the FSAR.

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

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

Response: No.

The proposed amendment involves changes to the Technical Specifications to properly reflect the surveillance requirements of City Water Tank. The proposed change provides assurance of availability of adequate volume of water in the CWT to cool the RCS from 102% rated thermal power to RHR entry conditions in 10 hours, if the CST is unavailable or depleted for any reason, and verifies the correct position of isolation valves in the flow path between the CWT and the AFS pump suction. These changes do not affect any accident initiators.

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

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

Response: No.

The proposed amendment involves changes to the Technical Specifications to properly reflect the surveillance requirements of City Water Tank. The proposed change to verify the CWT volume would ensure that an adequate volume of CW is available in the tank to cool the RCS from 102% rated thermal power to RHR entry conditions in 10 hours, if the CST is unavailable or depleted for any reason. The proposed change to verify the valve position for isolation valves in the flow path between the CWT and the AFS pump suction would ensure that isolation valves in the flow path are properly positioned. The proposed amendment does not involve any changes to plant equipment, or the way in which the plant is operated.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 26, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.6.5.b, ''Core Operating Limits Report (COLR),'' to incorporate the reference to Westinghouse topical report WCAP–12945–P–A, ''Code Qualification Document for Best Estimate Loss-of-Coolant Analysis [LOCA],'' dated March 1998. The proposed amendment would also allow the use of the analytical methodology to determine the core operating limits.

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.

No physical changes are being made by this change. The proposed changes involve use of the Best Estimate Large Break LOCA [loss-of-coolant accident] analysis methodology and associated TS [technical specification] changes. The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant. Therefore, there will be no increase in the probability of a loss of coolant accident. The consequences of a LOCA are not being increased. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46 paragraph b, that is it meets the five criteria listed in Section II of this evaluation. No other accident is potentially affected by this change.

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 previously analyzed?

Response: No.

There are no physical changes being made to the plant. No new modes of plant operation are being introduced. The parameters assumed in the analysis are within the design limits of existing plant equipment. All plant systems will perform equally during the response to a potential accident.

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

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

Response: No.

It has been shown that the analytic technique used in the analysis more realistically describes the expected behavior of the Indian Point 3 reactor system during a postulated loss of coolant accident. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of loss of coolant accidents with different break sizes, different locations and other variations in properties have been analyzed to provide assurance that the most severe postulated loss of coolant accidents were calculated. It has been shown by the analysis that there is a high level of probability that all criteria contained in 10 CFR 50.46 paragraph b) are met.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 26, 2002.

Description of amendment request: Extend the use of the pressure-temperature (P–T) limits in Technical Specification (TS) Figure 3.4.6.1–1 to 32 effective full power years by deleting a note on each unit's TS Figure limiting the validity of the Figure.

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

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

No. The proposed change to the technical specifications to extend the use of the existing pressure-temperature (P-T) limits does not affect the operation or configuration of any plant equipment. Thus, no new accident initiators are created by this change. The existing P-T limits are based on the projected reactor vessel neutron fluence at 32 effective full power years (EFPY) of operation specified in the current licensing basis for LGS [Limerick Generating Station], Units 1 and 2. A plant-specific calculation of reactor vessel 32 EFPY fast neutron fluence has been completed for LGS, Units 1 and 2, using the methodology described in a General Electric (GE) Company Licensing Topical Report (LTR), which adheres to the guidance in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The three-dimensional spatial distribution of neutron flux was modeled by combining the results of two separate two-dimensional neutron transport calculations. The latest available cross section libraries for the important components of Boiling Water Reactor (BWR) neutron flux calculations, i.e., oxygen, hydrogen and individual iron isotopes, were included. The resulting reactor vessel fast neutron fluence value is lower than the value in the current licensing basis for LGS, Units 1 and 2. Therefore, the existing 32 EFPY P-T limits bound the fast neutron fluence value calculated using the GE methodology. This provides sufficient assurance that the LGS, Unit 1 and Unit 2, reactor vessels will be operated in a manner that will protect them from brittle fracture under all operating conditions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed change to the technical specifications to extend the use of the existing P–T limits does not affect the operation or configuration of any plant equipment. The current P–T limits will remain valid and conservative during the proposed extension. 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?

No. The proposed change extends the use of the existing P–T limits. The existing P–T limits are based on the projected reactor vessel neutron fluence at 32 EFPY of operation specified in the current licensing basis for LGS, Units 1 and 2. A plant-specific calculation of reactor vessel 32 EFPY fast neutron fluence has been completed for LGS, Units1 and 2, using the NRC [Nuclear Regulatory Commission] approved methodology in a GE LTR, which adheres to the guidance in Regulatory Guide 1.190. The three-dimensional spatial distribution of neutron flux was modeled by combining the results of two separate two-dimensional neutron transport calculations. The latest available cross section libraries for the important components of BWR neutron flux calculations, i.e., oxygen, hydrogen and individual iron isotopes, were included. The resulting reactor vessel fast neutron fluence value is lower than the value in the current licensing basis for LGS, Units 1 and 2. Therefore, the existing 32 EFPY P–T limits bound the fast neutron fluence value calculated using the GE methodology. This provides sufficient margin such that the LGS, Unit 1 and Unit 2, reactor vessels will be operated in a manner that will protect them from brittle fracture under all operating conditions. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Acting Section Chief: Jacob I. Zimmerman.

Exelon Generation Company, LLC, and PSEG Nuclear LLC, Docket No. 50–277, Peach Bottom Atomic Power Station, Unit 2, York County, Pennsylvania

Date of application for amendment: June 10, 2002

Description of amendment request: Exelon Generation Company, LLC, the licensee, is proposing a change to the Peach Bottom Atomic Power Station (PBAPS), Unit 2, Technical Specifications (TSs) contained in Appendix A to the Operating License. This proposed change will revise the TS section on safety limits to incorporate revised safety limit minimum critical power ratios (SLMCPRs) due to the cycle-specific analysis performed by Global Nuclear Fuel for PBAPS, Unit 2, Cycle 15.

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

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

The derivation of the cycle specific safety limit minimum critical power ratios (SLMCPRs) for incorporation into the (TS[s]), and their use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Amendment 25 was approved by the NRC in a March 11, 1999 safety evaluation report.

The basis of the SLMCPR calculation is to ensure that greater than 99.9% of all fuel rods in the core avoid transition boiling if the limit is not violated. The new SLMCPRs preserve the existing margin to transition boiling. The GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which provides the fuel licensing acceptance criteria. The probability of fuel damage will not be increased as a result of this change. Therefore, the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The SLMCPR is a TS numerical value, calculated to ensure that transition boiling does not occur in 99.9% of all fuel rods in the core if the limit is not violated. The new SLMCPRs are calculated using NRC approved methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. Additionally, the GE–14 fuel is in compliance with Amendment 22 to ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U. S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which provides the fuel licensing acceptance criteria. The SLMCPR is not an accident initiator, and its revision will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

There is no significant reduction in the margin of safety previously approved by the NRC as a result of the proposed change to the SLMCPRs, which includes the use of GE–14 fuel. The new SLMCPRs are calculated using methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–14 (GESTAR–II), and U.S. Supplement, NEDE–24011–P–A–14–US, June, 2000, which incorporates Amendment 25. The SLMCPRs ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated when all uncertainties are considered, thereby preserving the fuel cladding integrity. Therefore, the proposed TS change will not involve a significant reduction in the margin of safety previously approved by the NRC.

Based on the above, Exelon Generation Company, LLC, concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Jacob I. Zimmerman, Acting.

Florida Power and Light Company, et al., Docket Nos. 50–335 and 50–389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: July 18, 2002

Description of amendment request: The proposed amendments would implement an administrative change to relocate the Technical Specifications (TS) requirements for the spent fuel crane to the respective unit's Updated Final Safety Analysis Report (UFSAR).

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

1. Would operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes to the Technical Specifications are administrative in nature in that the Technical Specifications for operation and surveillance of the spent fuel cask crane and the fuel handling crane will be relocated from Appendix A of the facility operating license to the UFSAR for each unit. The crane operation and surveillance requirements are not altered by this relocation. Once relocated, any future changes will be controlled by 10 CFR 50.59, and the UFSARs will be updated pursuant to 10 CFR 50.71(e). Because no operating requirements are changed by the proposed amendment, crane operation following the proposed amendment would not differ from current crane operation. The proposed Technical Specification changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do the changes alter any assumptions or conditions in any of the plant accident analyses. Therefore, facility operation in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR.

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

No. The proposed amendment will not affect the design function of any system, structure, or component. Relocating the existing Technical Specification requirements for the spent fuel cask crane and the fuel handling crane to the UFSAR is an administrative change and will not modify the physical plant or the modes of plant operation defined in the Facility Operating License. The operating restrictions imposed on the spent fuel-related cranes by the existing Technical Specifications will be retained in the UFSAR under this change. The change does not involve the addition or modification of equipment, nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different accident from any accident previously evaluated.

3. Would operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?

No. The proposed changes to the Technical Specifications are administrative in nature in that the Technical Specifications for operation and surveillance of the spent fuel cask crane and the fuel handling crane will be relocated from Appendix A of the facility operating license to the UFSAR for each unit. The crane operating restrictions that are being relocated to the UFSAR by this change are not being relaxed or eliminated. The proposed changes do not alter the basis for any technical specification that is related to the establishment of or the maintenance of a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety as defined in the basis for any Technical Specification or in any licensing document.

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

Date of amendment request: July 3, 2002.

Description of amendment request: The amendment would revise the Improved Technical Specifications (ITS) 3.8.1 and associated bases, ''AC Sources—Operating,'' by extending the allowed outage time for the emergency diesel generators (EDGs) from 72 hours to 14 days and to modify a note for two EDG ITS Surveillance Requirements (SRs).

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

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

The proposed license amendment extends the Completion Time for restoring an inoperable EDG to OPERABLE status and permits performance of certain SRs at power under specified conditions. The EDGs are designed to supply backup AC power to equipment in essential safety systems in the event of a loss of offsite power, and as such, the EDGs are not initiators of any design basis accident.

The design functions, operational characteristics, and interfaces between the EDGs and other plant systems will not be affected by the change. In addition, the initial conditions and assumptions for accidents that require the EDGs will remain unchanged. Defense in depth will be maintained by the redundant OPERABLE EDG, diverse 1E offsite power sources, and the availability of multiple emergency feedwater (EFW) and auxiliary feedwater (AFW) equipment capable of operating independently of both offsite power and the EDGs.

A Probabilistic Safety Assessment (PSA) has been performed to quantitatively assess the risk impact of an increase in Completion Times. Although the proposed changes result in slight increases in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP), and large early release frequency (LERF) and incremental conditional large early release probability (ICLERP), these increases are well below values that are considered risk significant in accordance with current regulatory guidance.

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

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

The proposed amendment extends the Completion Time for restoring an inoperable EDG to OPERABLE status and permits performance of certain SRs at power under specified conditions. The proposed amendment will not result in changes to the design, physical configuration or operation of the plant or the assumptions made in the safety analysis for accidents that require the EDGs. In addition, the proposed amendment will not result in changes to corrective or preventive maintenance activities associated with the EDGs, plant operating procedures, or the procedures used to respond to abnormal or emergency conditions. Assumptions made in the safety analysis related to EDG availability will also remain unchanged. Performance of certain SRs at power requires an evaluation to assure plant safety is maintained or enhanced, which would include evaluation for new or different plant conditions. As such, no new failure modes are being introduced. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed license amendment increases the Completion Times for restoring an inoperable EDG to OPERABLE status and permits performance of certain SRs at power under specified conditions. The proposed changes will improve EDG reliability by providing flexibility in scheduling and performing EDG preventive and corrective maintenance activities. This flexibility will reduce the probability (and associated risk) of a plant shutdown to repair an inoperable EDG that cannot be restored within the current ITS 3.8.1 Completion Times. Performance of the proposed SRs at power requires an evaluation to assure plant safety is maintained or enhanced. The proposed change will also increase the availability of the EDGs during MODE 5 and 6 outages, thus reducing shutdown risk.

The proposed amendment will not change the plant design, safety analysis, or the design, configuration or operation of the EDGs. The EDGs are designed to supply backup AC power to equipment in essential safety systems in the event of a loss of offsite power. Either EDG is capable of performing this function; therefore, as long as one train is available, the margin of safety is maintained. Defense in depth will be provided by the redundant OPERABLE EDG, the availability of diverse offsite circuits capable of supplying power to plant emergency loads, and EFW and AFW equipment that can perform their design function independently of both offsite power and the EDGs.

To ensure these defense in depth capabilities are maintained during required EDG maintenance, maintenance and surveillance activities that have the ability to impact the availability of the redundant EDG, required support systems and/or backup systems, the EFW and AFW systems and the 1E offsite power circuits will be controlled in accordance with the normal work controls process. As part of this process, weekly qualitative and quantitative risk assessments of scheduled on-line maintenance activities, and additional risk assessments of emergent work activities, will be performed in accordance with the guidance provided in CR–3 Compliance Procedure CP–253, ''Power Operation Risk Assessment and Management.'' If the results of these assessments indicate an increase in risk, appropriate actions to control temporary and aggregate risk increases and minimize risk increases above the overall plant baseline will be implemented in accordance with CP–253.

Additional measures to minimize risk will include increased administrative controls related to switchyard access, and increased inspection of identified risk significant fire areas within the plant. A Tier 2 analysis has also been performed to identify the dominant risk significant plant configurations during the time that an EDG is inoperable due to required corrective or preventive maintenance, and appropriate configuration controls/restrictions will be established prior to extended EDG maintenance.

As discussed in question (1) above and in the submittal, the slight increases in CDF, ICCDP, LERF and ICLERP resulting from the proposed amendment are all below values that are considered risk significant in accordance with the guidance provided in Regulatory Guide 1.174, ''An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,'' for changes to the plant, and Regulatory Guide 1.177, ''An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,'' for proposed increases in ITS Completion Times.

Based on the above, this 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Acting Section Chief: Kahtan N. Jabbour.

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

Date of amendment request: June 13, 2002.

Description of amendment request: The proposed technical specifications change request (TSCR) No. 79, Revision 1, is to revise Three Mile Island Nuclear Generating Station, Unit 2 (TMI–2) Technical Specification (TS) Administrative Controls section that will provide consistency with Three Mile Island Nuclear Generating Station, Unit 1, (TMI–1) TS changes submitted by AmerGen Energy Company, LLC (AmerGen) and Exelon Generation Company, LLC (EGC), which are currently under review by the U.S. Nuclear Regulatory Commission (NRC). GPU Nuclear utilizes EGC/AmerGen administrative controls under contract to TMI–2. The proposed request would delete TS Sections 6.4, ''Training,'' and 6.5.4, ''Independent Onsite Safety Review Group'' (IOSRG) from the administrative requirements in Section 6 of the TMI–2 Post Defueled Monitored Storage (PDMS) TS. Additionally, the IOSRG has been removed from the list of recipients of audit reports in Section 6.5.3.2.

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

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

TMI–2 is a defueled facility holding a Possession Only License is being maintained in Post Defueling Monitored Storage (PDMS). The introduction of the PDMS Quality Assurance Plan states in part in the second paragraph, ''Since the plant will be in a non-operating and defueled status, there will no longer be any structures, systems, or components that perform a safety function.''

Deletion of the technical specifications requirements for training and the IOSRG will have no adverse effect on any plant system; will not alter the source term, containment isolation, or allowable radiological consequences. These administrative changes will have no effect on any plant systems, structures or components and do not affect the physical plant, operating procedures, maintenance procedures, or emergency procedures at TMI–2.

The elimination of the IOSRG oversight function removes a function that is redundant to other oversight programs, not required by NRC regulation, and is not needed for the safe monitoring of TMI–2. Programmatic assessments of the TMI–2 programs will continue to be assessed by Nuclear Oversight personnel in accordance with the PDMS Quality Assurance Plan. Training will continue to be conducted in accordance with regulatory requirements.

The training programs for appropriate unit staff personnel other than licensed operators is now addressed by 10 CFR 50.120. With the 10 CFR 50.120 rule, the NRC is emphasizing the need to ensure that industry personnel training programs are based upon job performance requirements. This will be accomplished using the systems approach to training implemented by INPO [Institute of Nuclear Power Operations] accredited training programs for selected nuclear personnel. Included within the rule is the requirement that the training program must reflect industry experience. Deletion of the training requirements in the technical specifications will conform the license to the current requirements of 10 CFR 50.120.

Therefore, these changes will 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?

These changes are administrative in nature and do not affect any system functional requirements, plant maintenance, or operability requirements. The proposed changes involve the elimination of a redundant oversight function and the replacement of training requirements by the more vigorous requirements of 10 CFR 50.120, which are applicable to operating plants.

The proposed changes have no direct effect on any plant systems or components. The programs for the monitoring, surveillance, or maintenance of TMI–2 are unaffected. Oversight of TMI–2 will continue to be provided by Nuclear Oversight personnel and the TMI–2 Safety Oversight Committee in accordance with the requirements of the PDMS Quality Assurance Plan.

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

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

The training and IOSRG requirements contained in TMI–2 Technical Specifications Section 6.0 ''Administrative Controls'' are administrative in nature. The proposed changes have no direct effect on any plant systems. There are currently no safety limits that apply to TMI–2 during PDMS. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review it appears that the three standards of 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 & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: June 28, 2002.

Description of amendment request: The licensee proposed changes to surveillance requirements in Table 4.6.2b, ''Instrumentation that Initiates Primary Coolant System or Containment Isolation,'' of the Nine Mile Point Nuclear Station, Unit No. 1 (NMP1) Technical Specifications (TS) regarding the isolation capability of the shutdown cooling system (SDCS). Specifically, the changes will remove the restriction to perform channel functional testing and channel calibration associated with SDCS high area temperature only during refueling outages. The changes will allow these surveillance activities to be performed during other operating conditions on a once-per-operating-cycle basis, thereby maintaining SDCS availability to support reactor shutdown operations during refueling.

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 Nine Mile Point Unit 1 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The only safety-related functions of the SDCS are (i) to maintain the integrity of the reactor coolant pressure boundary, and (ii) to provide primary containment isolation of the shutdown cooling lines. The proposed amendment removes an unnecessary restriction to perform channel functional testing and calibration associated with SDCS isolation capability only during refueling outages. It provides the flexibility to perform these surveillances during other operating conditions on a ''once per operating cycle'' basis. The change does not modify the surveillance frequency, surveillance acceptance criteria, high area temperature setpoint limit for initiating SDCS isolation, plant equipment configurations during SDCS surveillances, or the existing requirements for maintaining SDCS isolation and reactor coolant pressure boundary integrity.

Based on the above, the operation of NMP1 in accordance with the proposed amendment will not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 1 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve any physical modifications to the plant and does not alter equipment configuration, setpoints, safety parameters, surveillance interval durations, or surveillance acceptance criteria. It does not affect the operation of any safety-related structure, system, or component in a manner that could introduce a new accident precursor or a new failure mechanism. The SDCS isolation valves will continue to perform their isolation function by remaining closed with power removed during power operation of the reactor.

Based on the above, the operation of NMP1 in accordance with the proposed amendment cannot create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 1 in accordance with the proposed

amendment will not involve a significant reduction in a margin of safety.

The proposed change does not affect any of the plant's fission product barriers or safety/operational limits. The high area temperature setpoint for SDCS isolation will remain within the existing TS limit.

The SDCS isolation valves will continue to remain closed with power removed during power operation of the reactor. The proposed ''[o]nce per operating cycle'' surveillances will be adequate to ensure acceptable SDCS equipment operability and reliability.

Based on the above, the operation of NMP1 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: July 12, 2002.

Description of amendment request: The licensee proposed to change the Technical Specifications (TSs), Sections 3.1.1 and 4.1.1, ''Control Rod System,'' by reducing the power level below which the rod worth minimizer (RWM) or a second independent verification of rod positions must be used from 20% rated thermal power (RTP) to 10% RTP. The licensee stated that analysis has shown that no significant control rod drop accident (CRDA) can occur above 10% RTP. The low power setpoint change will reduce the time necessary for both reactor startup and shutdown.

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 reproduced below:

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The TS revision lowers the power level at which the analyzed rod position sequence must be followed by use of the RWM or a second independent verification of rod positions. The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. Compliance with the analyzed rod position sequence and operability of the RWM is required in the startup and run modes when thermal power is less than 10% RTP. When thermal power is 10% RTP or greater, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gram fuel design limit during a CRDA. None of the accidents previously evaluated assume the RWM is an initiator of the accident and therefore, the probability of an accident is not significantly increased by the change. Because the fuel design limit is not exceeded, the change to the low power setpoint will not significantly increase the consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The TS revision lowers the power level below which the analyzed rod position sequence must be followed. The change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, a new or different type of accident from any accident previously evaluated is not created.

The operation of Nine Mile Point Unit 1, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The RWM enforces the analyzed rod position sequence to ensure that the initial conditions of the CRDA analysis are not violated. Compliance with the analyzed rod position sequence and operability of the RWM are required in the startup and run modes when thermal power is less than 10% RTP. When thermal power is 10% RTP and greater, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gram fuel design limit during a CRDA. Because the fuel design limit is not exceeded at 10% RTP and greater, the change to the RWM low power setpoint does not significantly reduce the margin of safety.

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

Attorney for licensee: 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: July 12, 2002.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) Section 3.1.a.3, ''Pressurizer Safety Valves.'' Also, the proposed amendment would reformat TS 3.1.a.3 to more closely resemble the format of Improved Standard Technical Specification (ISTS) to improve clarity. The proposed amendment would allow both pressurizer safety valves to be inoperable or removed while the reactor vessel head is on. This would only be applicable when the temperature and pressure are low enough such that the Low Temperature Overpressure Protection (LTOP) System can safely protect the Reactor Coolant System (RCS). The TSs currently requires the LTOP System to protect the RCS when the RCS temperature is less than LTOP enabling temperature.

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 format changes are administrative in nature and therefore have no effect on the probability or consequences of an accident. The situation where the plant has two inoperable or removed pressurizer safeties while the LTOP System is enabled is not considered an accident initiator. Therefore, any change to the system would not affect the probability of an accident previously evaluated. The risk of core damage/release of radioactivity would not increase with all of the other plant safety features still in place.

The proposed changes adds clarity to the TSs by describing a specific situation when the RCS is at low temperature & pressure while overpressure protection is provided by the LTOP System. Since this TS change is not an accident initiator and existing TS will ensure the LTOP System will continue to protect the RCS pressure boundary, this proposed amendment does not involve an increase in 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 situation where the plant has two inoperable pressurizer safeties while the LTOP System is enabled is not considered an accident initiator. A failure of this system will not result in an accident. The format changes are administrative in nature and therefore have no effect on the probability or consequences of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a change to the physical plant or operations. As the RCS temperature is lowered to less than 200 °F, the LTOP System provides the RCS overpressure protection required. Since the LTOP System is currently approved for use by TS 3.1.b.4, it would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, any change to the system would not affect the probability of an accident previously evaluated.

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

The format changes are administrative in nature and therefore are not involved in a significant reduction in the margin of safety. Margin of safety relates to overpressure protection when the RCS is less than 200 °F. This margin is controlled by the LTOP System completely and does not rely on the pressurizer safeties. This proposed amendment allows KNPP to have both pressurizer safeties to be inoperable as long as the RCS is below the LTOP System enabling temperature. Therefore, NMC concludes that there is not a significant reduction in the margin of safety.

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

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

NRC Section Chief: L. Raghavan.

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

Date of amendment request: June 24, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.5.3, ''Post Accident Sampling System (PASS),'' to eliminate the requirements to have and maintain the PASS at Plant Hatch. 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 application 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 June 24, 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 [Three Mile Island, Unit 2] 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 amendment request, the requested change does not involve a significant hazards consideration.

The NRC staff proposes 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.

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

Date of amendment request: July 11, 2002.

Description of amendment request: The proposed amendments would delete Technical Specification 3.3.1.1.I.2, which requires returning the Oscillating Power Range Monitor to operable status within 120 days of discovering its operability.

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 Oscillating Power Range Monitor (OPRM) is not designed for the prevention of an instability event or any other previously evaluated event. Accordingly, it cannot increase the probability of an instability event or any other previously evaluated event.

The consequences of the instability event are not significantly increased, because the alternate method of detection and suppression of thermal-hydraulic instability oscillations is well established at Plant Hatch. Furthermore, operators are adequately trained on instabilities.

This proposed change to delete the 120-day Completion Time restriction on an inoperable OPRM does not affect any other system designed for the mitigation of previously analyzed events.

For the above reasons, the probability and consequences of a previously analyzed event are not increased.

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

The proposed change only deletes a Technical Specification requirement. It does not physically alter the design, operation, testing, or maintenance of any plant system or piece of equipment. The proposed change introduces no new modes of operation. Consequently, the change does not create the possibility of a new or different kind [of] event.

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

The proposed change deletes the requirement to restore the OPRM system to operable status within 120 days of discovering its inoperability. A manual alternate method to detect and suppress thermal-hydraulic instability oscillations has been included in Plant Hatch procedures for many years. Also, operators are trained on instability events.

Accordingly, the manual alternate method is adequate and thus, the margin of safety for the instability event 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.

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.

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: August 2, 2001.

Description of amendment request: The proposed amendment revises Technical Specifications to extend, on a one-time basis, the current interval for Type A testing from 10 years to 15 years.

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

The proposed Technical Specification revision extends the current interval for Type A testing. The current test interval of ten years would be extended on a one-time basis to 15 years from the preceding Type A test. Pursuant to 10 CFR 50.91, this analysis provides a determination that the proposed change to the Technical Specifications for a one-time extension of the interval for Integrated Leakage Rate Testing does not involve any significant hazards consideration as defined in 10 CFR 50.92.

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

The proposed extension to the Type A testing interval will not increase the probability of an accident previously evaluated. The containment Type A testing interval extension is not a modification and the testing interval extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to the Type A testing interval does not involve a significant increase in the consequences of an accident. Research documented in NUREG–1493 has determined that Type B and C tests can identify the vast majority (more than 95%) of all potential leakage paths.

NUREG–1493 concluded that reducing the Type A test frequency to one per twenty years leads to an imperceptible increase in risk. Testing and inspection provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. Previous Type A tests show leakage does not exceed acceptance criteria, indicating a very leak-tight containment. Inspections required by the Maintenance Rule and ASME code are performed in order to identify indications of containment degradation that could affect leak tightness.

Experience at the South Texas Project demonstrates that excessive containment leakage paths are detected by Type B and C Local Leakage Rate Tests. Type B and C testing will identify any containment opening, such as a valve, that would otherwise be detected by the Type A tests. These factors show that a Type A test interval extension will not involve a significant increase in the consequences of an accident.

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

The proposed extension of the Type A testing interval will not create the possibility of a new or different type of accident from any previously evaluated. There are no physical changes being made to the plant and there are no changes in operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

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

The proposed extension of the Type A testing interval will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing results in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have a minimal effect on this risk because 95% of the potential leakage paths are detected by Type B and C testing.

Deferral of Type A testing for the South Texas Project does not increase the level of public risk due to loss of capability to detect and measure containment leakage or loss of containment structural capability. Other containment testing methods and inspections will assure all limiting conditions of operation will continue to be met. The margin of safety inherent in existing accident analyses is maintained.

Based on the evaluation provided above, the South Texas Project concludes that the proposed change does not involve a significant hazards consideration and will not have a significant effect on safe operation of the plant. Therefore, there is reasonable assurance that operation of the South Texas Project in accordance with the proposed revised Technical Specifications will not endanger the public health and safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 10, 2002.

Description of amendment request: The proposed one-time technical specification (TS) change revises the Sequoyah Unit 2 Limiting Condition for Operation for Section TS 3.7.4, ''Essential Raw Cooling Water System,'' to include provisions for maintaining operability of this system during performance of heavy load lifts associated with the Unit 1 steam generator replacement (SGR) project. The provisions should ensure safe operation of Unit 2 during heavy load lift activities. In addition, compensatory measures proposed should ensure safe shutdown capability of Unit 2 in the unlikely event a heavy load drop occurs over Essential Raw Cooling Water system piping.

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

TVA has concluded that operation of Sequoyah (SQN) Unit 2, in accordance with the proposed change to Technical Specification (TS) 3/4.7.4, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

TVA's proposed license amendment is a one-time change to the SQN Unit 2 TSs. The proposed change revises SQN Limiting Condition for Operation 3.7.4, ''Essential Raw Cooling Water System,'' to include provisions for maintaining operability of this system during performance of heavy load lifts associated with the Unit 1 steam generator replacement (SGR) project. The provisions ensure safe operation of Unit 2 during heavy load lift activities. In addition, compensatory measures ensure safe shutdown capability of Unit 2 in the unlikely event a heavy load drop occurs over ERCW system piping.

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

No changes in event classification as discussed in SQN Updated Final Safety Analysis Chapter 15 will occur due to the proposed TS amendment. The one-time TS provision ensures that the SQN essential raw cooling water (ERCW) system remains operable for continued safe operation of Unit 2 during heavy load lifts performed on Unit 1 during SGR replacement activities.

Accordingly, the proposed modification to SQN Unit 2 TSs and the implementation of compensatory measures for a postulated load drop will not significantly increase the probability or consequences of an accident previously evaluated.

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

The possibility of a new or different accident scenario occurring as a result of activities conducted during the SQN Unit 1 SGR project are not created. Three postulated scenarios related to heavy load handling during the SGR project were examined for their potential to represent a new or different kind of accident from those previously evaluated: (1) a breach of the old steam generator (OSG), resulting in the release of contained radioactive material, (2) flooding in the Auxiliary Building caused by the failure of piping in the ERCW tunnel, and (3) loss of ERCW to support safe shutdown of the operating unit.

Failure of an OSG that results in a breach of the primary side of the steam generator (SG) could potentially result in a release of a contained source outside containment. The consequences of this event, both offsite and in the control room, were examined and found to be within the consequences of the failure of other contained sources outside containment at the SQN site (*i.e.,* within the SQN design basis).

With regard to flooding of the Auxiliary Building from a heavy load drop, the protective measure taken prior to the lifting of heavy loads include installation of a wall in the ERCW tunnel near the Auxiliary Building interface. The wall provides protection against a postulated flood of the ERCW tunnel and protects against flooding of the Auxiliary Building beyond those events previously evaluated.

With regard to the potential for a heavy load drop causing the loss of ERCW cooling water to the operating unit (i.e., Unit 2), TVA is implementing provisions to preclude a load drop. A heavy load drop is considered an unlikely accident for the following reasons:

- The lifting equipment was specifically designed and chosen for the subject heavy lifts,
- Crane operators will be specially trained in the operation of the lift equipment and in the SQN site conditions,
- Qualifying analyses and administrative controls will be used to protect the lifts from the effects of external events,
- The areas over which a load drop could cause loss of ERCW are a small part of the total travel path of the loads.

In addition, protection against the potential for a loss of ERCW is established prior to any heavy load lifts. Compensatory measures ensure the ERCW system is isolated should a pipe break occur, and that ERCW flow is redirected to equipment essential for safe shutdown capability of Unit 2.

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

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

The proposed change to the Unit 2 TSs support safe operation and safe shutdown capability of Unit 2 during replacement of the Unit 1 SGs. These measures do not result in changes in the design basis for plant structures, systems, and components (SSCs). Consequently, the proposed change will not affect any margins of safety for plant SSCs.

Accordingly, a significant reduction in the margin of safety is not created by the proposed change.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

Date of application for amendments: March 29, 2002 (TS 02–02).

Brief description of amendments: The proposed amendment would change the Sequoyah (SQN) Unit 1 Technical Specifications (TSs) by revising Specification 3/4.4.5 to eliminate surveillance requirements associated with two alternate repair criteria. The associated License Condition 2.C.9.d is also deleted. In addition, the proposed change revises SR 3/4.4.5.3.a to allow a one-time, 40-month steam generator (SG) inspection interval after the first (post-Unit 1 SG replacement) inservice inspection resulting in a C–1 category. The proposed change is in lieu of the current TS criteria that requires two consecutive category C–1 inspections for application of the 40-month SG inspection interval.

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

TVA has concluded that operation of Sequoyah Nuclear Plant (SQN) Unit 1, in accordance with the proposed change to the technical specifications and License Condition, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

TVA is proposing to modify SQN Unit 1 TS 3/4.4.5, ''Steam Generators'' to delete surveillance requirements (SRs) that describe steam generator (SG) tube plugging limits for two alternate repair criteria (ARC). The first ARC is for axial outside diameter stress corrosion cracking (ODSCC) at non-dented tube support plates and the second ARC is for axial primary water stress corrosion cracking (PWSCC) at dented tube support plates. TVA's proposed amendment removes both ARCs through the deletion of the following SRs: SR 4.4.5.2.b.4, 4.4.5.2.d, 4.4.5.2.e, a portion of 4.4.5.4.a.6, 4.4.5.4.a.10, 4.4.5.4.a.11, 4.4.5.5.d, and 4.4.5.5.e. TVA's proposed removal of these SRs for ARC reestablishes standard tube plugging criteria within the TS for SQN Unit 1. Returning to the standard TS 40 percent through-wall tube plugging limit is inherently more conservative.

Included with the above change is deletion of License Condition 2.C.9.d that references prior TVA commitment letters for SG inspection. The TVA letters and their commitments will no longer apply following replacement of the Unit 1 SGs.

In addition, TVA is proposing a revision to TS 3/4.4.5.3.a to allow application of the 40-month inspection interval after one SG inspection resulting in a C–1 category. The proposed change replaces the current TS requirement that invokes the extended 40-month inspection interval after two consecutive inspections resulting in a category of C–1. TVA's proposed change provides a relaxation of the SG inspection requirements and schedule. The relaxation in the inspection schedule is intended to coincide with replacement of SQN Unit 1 SGs during the Cycle 12 refueling outage (Spring 2003). The replacement of the SQN Unit 1 SGs incorporate significant design improvements that include thermally treated Alloy 690 SG tubing. The improvements in SG design and tube material properties increase the resistance to SG tube degradation mechanisms and allow optimization of SG inspection schedules. The proposed optimization of SG inspections reduce the cumulative number of SG inspections over the life of the plant and result in significant dose, schedule, and cost savings to TVA.

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

TVA's proposed TS amendment does not compromise limits associated with SG tube integrity. TVA's proposed change removes existing SG tube plugging criteria (*i.e.*, ARC) from the TS and reestablishes the standard TS criteria (40 percent through-wall criteria). This change is inherently more conservative. The proposed allowance for an extended inspection interval is a conservative inspection strategy that is based on improved SG design features and SG tube materials that have been shown to resist degradation and preserve SG tube integrity.

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

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

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

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

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

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

TVA's proposed change to include a one-time extension to the SQN Unit 1 SG inspection interval retains conservative inspection strategy that maintains the structural and leakage integrity of the SGs. TVA intends to replace SQN Unit 1 SGs during the Cycle 12 refueling outage and perform a 100 percent full length inspection of SG tubes during the Cycle 13 refueling outage to verify that damage mechanisms do not exist. Twelve years of SG operation history indicate that corrosion damage mechanisms do not appear in replacement SGs that contain thermally treated Alloy 690 tubing. The replacement SG design also contains design improvements that provide reasonable assurance that tube degradation is not likely to occur over the proposed 40-month operating period (Cycle 13 refueling outage to Cycle 15 refueling outage). The corrosion resistant properties of the thermally treated Alloy 690 tubing and the improved design will limit the initiation of damage mechanisms and limit growth rate such that tube structural and leakage integrity will be maintained over two operating cycles.

TVA's proposed change to extend the SG inspection interval does not result in a change to system design features. The proposed change does not affect the plant conditions, setpoints, or safety limits that could result in precursors to accidents or degrade accident mitigation systems. Accordingly, plant system safety functions are not altered by the proposed change.

The effect of this change is to extend allowable SG inspection intervals while retaining conservative margins to maintain the structural and leakage integrity of the SGs. Consequently, the proposed TS revisions does not reduce the margin of safety.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

Date of application for amendments: July 10, 2002 (TS 01–09).

Brief description of amendments: The proposed amendment would change the Sequoyah (SQN) Unit 1 and 2 Technical Specifications (TSs) by removing the requirement to not make positive reactivity changes during certain conditions and replace it with requirements to maintain shutdown margin or boron concentration. The changes will permit limited positive reactivity changes that are necessitated by plant operations. These changes will limit the amount of reactivity changes to those that will continue to assure appropriate reactivity limits are met. The proposed changes are consistent with TS Task Force 286 and Revision 2 to NUREG–1431.

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

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

The proposed change does not involve an increase in the probability or consequences of an accident previously evaluated. The proposed activities to be allowed during certain operating conditions are permitted at other times during routine operating conditions. The changes do not affect the limits on reactivity that are specified in other specifications. The proposed changes continue to ensure restrictions on additions and flowpaths of unborated water that are in the existing specifications. The proposed change does not affect the limits on reactivity that are credited in the safety analysis. Therefore, no increase in the probability or consequences of any accident previously evaluated will occur.

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

The proposed changes permit the conduct of normal operating evolutions during limited periods when additional controls over reactivity margin are imposed by the TSs. The proposed change does not introduce any new equipment into the plant or significantly alter the manner in which existing equipment will be operated. The changes to operating allowances are minor and are only applicable during certain conditions. The operating allowances are consistent with those acceptable at other times. Since the proposed changes only allow activities that are presently approved and routinely conducted, no possibility exists for a new or different kind of accident from those previously evaluated.

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

The proposed changes do not involve a significant reduction in a margin of safety because the ability to make the reactor subcritical and maintain it subcritical during all operating conditions and modes of operation will be maintained. The margin of safety is defined by the shutdown margin limits and the refueling boron concentration limit. The proposed changes do not affect these operating restrictions and the margin of safety which assures the ability to make and maintain the reactor subcritical is not affected.

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

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

NRC Section Chief: Kahtan N. Jabbour, Acting.

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

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

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

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units 1 and 2, Louisa County, Virginia

Date of amendment request: July 18, 2002.

Brief description of amendment request: These amendments would revise the Facility Operating Licenses (FOLs) to change the implementation date for the Improved Technical Specifications (ITS), including the relocation of certain existing TS requirements to licensee-controlled documents, from no later than September 2, 2002, to no later than December 20, 2002.

*Date of publication of individual notice in **Federal Register:*** July 25, 2002 (67 FR 48679).

Expiration date of individual notice: August 26, 2002.

Notice of Issuance of Amendments to Facility 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 e—mail to pdr@nrc.gov.

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

Date of application for amendment: December 28, 2000, as supplemented May 31, 2002.

Brief description of amendment: The amendment decreases the allowed outage time for an inoperable channel or channels of the anticipated transient without scram recirculation pump trip instrumentation.

Date of issuance: July 17, 2002.

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

Amendment No.: 153.

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

*Date of initial notice in **Federal Register:*** February 7, 2001 (66 FR 9378). The supplemental letter did not significantly change the requested amendment or affect the proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 17, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 31, 2001, as supplemented by letters dated March 6, and April 23, 2002.

Brief description of amendment: The amendment revises License Condition 2.C.(3) of Operating License DPR–6 to reference revisions of the Big Rock Point Defueled Security Plan, Defueled Suitability Training and Qualification Plan, Defueled Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Plan.

Date of issuance: July 18, 2002.

Effective date: As of the date of issuance and shall be implemented prior to placing the spent fuel in the Big Rock Point Plant independent spent fuel storage installation.

Amendment No.: 123.

Facility Operating License No. DPR–6: The amendment revised the Facility Operating License.

Date of initial notice in **Federal Register:** August 22, 2001 (66 FR 44166). The March 6 and April 23, 2002, supplemental letters provided additional clarifying information that did not expand the scope of the application as originally noticed and did not change the NRC staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 18, 2002.

No significant hazards considerations comments received: No.

Dominion Nuclear Connecticut, Inc., et al., Docket No. 50–423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: February 5, 2002 as supplemented on March 6, 2002.

Brief description of amendment: The amendment changes the term in the technical specifications ''once each REFUELING INTERVAL'' to ''once per 24 months'' in several surveillance requirements.

Date of issuance: July 24, 2002.

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

Amendment No.: 206.

Facility Operating License No. NPF–49: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 28, 2002 (67 FR 36930). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 2002.

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: December 20, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 5.5.14 to eliminate the use of the term ''unreviewed safety question,'' and replace the word ''involve'' with the word ''require'' as it applies to changes made to the updated Final Safety Analysis Report and the TS Bases.

Date of issuance: July 17, 2002.

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

Amendment Nos.: 200 & 193.

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

Date of initial notice in ***Federal Register:*** March 5, 2002 (67 FR 10010). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 20, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) 5.5.14 to eliminate the use of the term ''unreviewed safety question,'' and replace the word ''involve'' with the word ''require'' as it applies to changes made to the updated Final Safety Analysis Report and the TS Bases.

Date of issuance: July 17, 2002.

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

Amendment Nos.: 204 & 185.

Facility Operating License Nos. NPF- and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2921). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 31, 2002, as supplemented by letter dated June 20, 2002.

Brief description of amendment: The amendment revises Technical Specification 3.8.1, ''AC Sources-Operating,'' to extend the allowed outage time for a Division 1 or Division 2 Diesel Generator from the current 72 hours to 14 days.

Date of issuance: July 16, 2002.

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

Amendment No: 151.

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

Date of initial notice in ***Federal Register:*** April 2, 2002 (67 FR 15623). The June 20, 2002, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 16, 2002.

No significant hazardous consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. 50–a254 and 50–265, Quad Cites Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: May 1, 2002.

Brief description of amendments: The amendments revise the start delay time in the surveillance for the emergency diesel generators from ''≤10 seconds'' to ''≤13 seconds.''

Date of issuance: July 17, 2002.

Effective date: For Unit 2, as of the date of issuance and shall be implemented within 30 days of the completion of Unit 1 refueling outage 17, which is scheduled for November 2002. For Unit 1, as of the date of issuance and shall be implemented within 30 days following the date when General Electric (GE)–14 fuel is loaded into the reactor, which is scheduled during refueling outage 17 in November 2002. The amendment may not be implemented prior to the date GE–14 fuel is loaded into the reactor.

Amendment Nos.: 206 and 202.

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

*Date of initial notice in **Federal Register:*** a May 28, 2002 (67 FR 36931). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 17, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 24, 2001, as supplemented June 5, and July 1.

Brief description of amendments: The amendments revise the Improved Technical Specifications (ITS) to accommodate future changes in plant design, including increased levels of Once—Through Steam Generator (OTSG) tube plugging. The changes are categorized into two sets. The first set of changes relocate parameters from the ITS to the cycle-specific Core Operating Limits Report (COLR). These parameters are the Variable Low Pressure Trip equation specified in ITS Table 3.3.1–1, and Reactor Coolant System (RCS) pressure limit within Surveillance Requirement (SR) 3.4.1.1. The second set of changes are applicable to raising the OTSG tube plugging limit to a maximum of 20% equivalent of all tubes, and addresses its impact. These changes include the revision of the hot leg maximum temperature limit, and the revision of the RCS minimum flow limits for four- and three-reactor coolant pump operation. The RCS limits associated with 20% tube plugging will be maintained in its ITS. Cycle-specific values of these limits, however, have been relocated to the COLR. The hot leg temperature and RCS flow limit values within SR 3.4.1.2 and 3.4.1.3 ''RCS Pressure, Temperature, and Flow DNB [departure from nucleate boiling] Limits,'' were relocated to reflect their location in the COLR. For both sets of changes, ITS 5.6.2.18(a) was modified to reflect the relocation of cycle-specific values from the ITS and the COLR.

Date of issuance: July 16, 2002.

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

Amendment Nos.: 204.

Facility Operating License Nos. DPR–72: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** a August 22, 2001 (66 FR 44173). The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 16, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: July 30, 2001, as supplemented by letter dated August 23, 2001.

Description of amendment request: The amendment revises the Cooper Nuclear Station's licensing basis.

Date of issuance: July 19, 2002.

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

Amendment No.: 192.

Facility Operating License No. DPR–46: Amendment revises the Cooper Nuclear Station's licensing basis.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. June 25, 2002 (67 FR 42828). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by July 29, 2002, but indicated that, if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated July 19, 2002.

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 20, 2002, as supplemented by letters dated June 19, July 3 (two letters), and July 12, 2002. The letters dated July 3 (two letters), and July 12, 2002, were of a clarifying nature, did not expand the application beyond the scope of the initial notice, and did not affect the staff's initial proposed no significant hazards consideration determination.

Description of amendment request: The amendment revises the Cooper Nuclear Station's Technical Specifications (TS) 3.7.2 and 3.7.3 reflecting increases in TS temperature limits for ultimate heat sink and reactor equipment cooling water temperatures.

Date of issuance: July 22, 2002.

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

Amendment No.: 193.

Facility Operating License No. DPR–46: Amendment revises the Cooper Nuclear Station's TS.

Public comments requested as to proposed no significant hazards consideration (NSHC): Yes. 67 FR 43688 dated June 28, 2002. The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided an opportunity to request a hearing by July 12, 2002, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendment, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a safety evaluation dated July 22, 2002.

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

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

Date of application for amendments: September 20, 2001, as supplemented by letters dated March 27 and April 12, 2002.

Brief description of amendments: The amendments revised the Technical Specifications to support extension of the operating cycle from 18 months to 24 months.

Date of issuance: July 12, 2002.

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

Amendment Nos.: 232/174.

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

*Date of initial notice in **Federal Register:*** November 28, 2001 (66 FR 59512). The supplements dated March 27 and April 12, 2002, provided clarifying information that did not change the scope of the September 20, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 12, 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: October 22, 2001, as supplemented by letters dated May 16 and June 25, 2002.

Brief description of amendments: The amendments change TS 3/4.9.4, ''Refueling Operations—Containment Building Penetrations'', to allow the equipment hatch to be open during core alterations or movement of irradiated fuel within the containment.

Date of issuance: July 18, 2002.

Effective date: July 18, 2002.

Amendment Nos.: Unit 1—139; Unit 2—128.

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 May 16 and June 25, 2002, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 18, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 26th day of July 2002.

For the Nuclear Regulatory Commission.

Ledyard B. Marsh,

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

[FR Doc. 02–19420 Filed 8–5–02; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

[Release No. IC–25690; File No. 812–12767]

American United Life Insurance Company, et al.; Notice of Application July 31, 2002.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of Application for an order pursuant to section 26(c) of the Investment Company Act of 1940, as amended (''1940 Act''), approving certain substitutions of securities, and pursuant to section 17(b) of the 1940 Act exempting related transactions from section 17(a) of the 1940 Act.

APPLICANTS: American United Life Insurance Company (''AUL''), AUL American Unit Trust (''AUL Account''), AUL American Individual Unit Trust (''AUL Individual Account'') and, with respect only to the relief requested pursuant to section 17(b), OneAmerica Funds, Inc. (''OneAmerica''). AUL, the AUL Account, the AUL Individual Account and OneAmerica are together referred to herein as the ''Applicants.''

SUMMARY OF APPLICATION: Applicants request an order to permit certain registered unit investment trusts to substitute (i) shares of common stock issued by OneAmerica Asset Director Portfolio (''Asset Director Portfolio''), a series of OneAmerica for shares of common stock issued by OneAmerica Tactical Asset Allocation Portfolio (''Tactical Asset Allocation Portfolio''), also a series of OneAmerica; and (ii) Investor Class shares issued by American Century Strategic Allocation: Conservative Fund (''Strategic Allocation: Conservative Fund''), American Century Strategic Allocation: Moderate Fund (''Strategic Allocation: Moderate Fund''), and American Century Strategic Allocation: Aggressive Fund (''Strategic Allocation: Aggressive Fund'' and, together with the Strategic Allocation: Conservative Fund and the Strategic Allocation: Moderate Fund, the ''Strategic Allocation Funds''), each a series of American Century Strategic Asset Allocations, Inc. (''American Century Strategic'') for shares of common stock issued by the OneAmerica Conservative Investor Portfolio (''Conservative Investor Portfolio''), OneAmerica Moderate Investor Portfolio (''Moderate Investor Portfolio''), and OneAmerica Aggressive Investor Portfolio (''Aggressive Investor Portfolio'' and, together with the Conservative Investor Portfolio and the Moderate Investor Portfolio, the ''Lifestyle Portfolios''), each a series of OneAmerica, respectively, currently held by those unit investment trusts, and to permit in-kind purchases and redemptions of portfolio securities in connection with the proposed substitution transactions relating to the Tactical Asset Allocation Portfolio and the Asset Director Portfolio (''In-Kind Transactions'').

FILING DATE: The application was filed on January 28, 2002, and amended and restated on July 26, 2002 (''Amended and Restated Application'').

HEARING OR NOTIFICATION OF HEARING: An order granting the Application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving Applicants with a copy of the request, personally or by mail. Hearing requests should be received by the SEC by 5:30 p.m. on August 26, 2002, and should be accompanied by proof of service on Applicants, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons who wish to be notified of a hearing may request notification by writing to the Secretary of the Commission.

ADDRESSES: Secretary, Securities and Exchange Commission, 450 Fifth Street, NW, Washington, DC 20549–0609. Applicants: c/o Richard A. Wacker, *Esq.*, American United Life Insurance Company, One American Square, Indianapolis, Indiana 46282. Copies to: Keith T. Robinson, Dechert, 1775 Eye Street, NW, Washington, DC 20006–2401.

FOR FURTHER INFORMATION CONTACT: Patrick F. Scott, Attorney, or Lorna J. MacLeod, Branch Chief, Office of Insurance Products, Division of Investment Management, at (202) 942–0670.

SUPPLEMENTARY INFORMATION: The following is a summary of the application; the complete application may be obtained for a fee from the Public Reference Branch of the Commission, 450 Fifth Street, NW, Washington, DC 20549, (202) 942–8090.

Applicants' Representations

1. AUL is an Indiana stock insurance company. AUL is the depositor and sponsor of the AUL Account and the AUL Individual Account, each a separate investment account established under Indiana law.

2. The AUL Account and the AUL Individual Account are each registered with the Commission under the 1940 Act as a unit investment trust. The assets of the AUL Account and the AUL Individual Account support certain individual and group variable annuity contracts (collectively, the ''Contracts''). The individual variable annuity contracts include Contracts for which premiums may vary in amount and frequency, subject to certain limitations and Contracts for which premiums may vary in amount and frequency during