III.

Accordingly, pursuant to sections 161b, 161i, 161o, and 184 of the Atomic Energy Act of 1954, as amended, 42 USC 2201(b), 2201(i), 2201(o), and 2234; and 10 CFR 50.80 and 10 CFR 72.50, it is hereby ordered that the application regarding the indirect license transfers referenced above is approved, subject to the following conditions:

(1) Following the completion of the indirect license transfers approved by this Order, PGE shall provide the Director of the Office of Nuclear Reactor Regulation and the Director of the Office of Nuclear Material Safety and Safeguards a copy of any application, at the time it is filed, to transfer (excluding grants of security interests or liens) from PGE to its parent, or to any other affiliated company, facilities for the production, transmission, or distribution of electric energy having a depreciated book value exceeding ten percent (10%) of PGE's consolidated net utility plant, as recorded on its books of account.

(2) Should the proposed stock purchase not be completed by March 31, 2003, this Order shall become null and void, provided, however, upon application and for good cause shown, such date may be extended.

This Order is effective upon issuance.

IV.

For further details with respect to this Order, see the initial application dated December 6, 2001, supplemental letter dated January 31, 2002, and the safety evaluation dated March 26, 2002, which 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, and 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*.

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

For the Nuclear Regulatory Commission.

E. William Brach,

Director, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards.

John A. Zwolinski,

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

[FR Doc. 02–7928 Filed 4–1–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Meeting Notice

AGENCY HOLDING THE MEETING: Nuclear Regulatory Commission.

DATE: Weeks of April 1, 8, 15, 22, 29, May 6, 2002.

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

STATUS: Public and Closed.

MATTERS TO BE CONSIDERED:

Week of April 1, 2002

There are no meetings scheduled for the Week of April 1, 2002.

Week of April 8, 2002—Tentative

Friday, April 12, 2002

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

Week of April 15, 2002—Tentative

There are no meetings scheduled for the Week of April 15, 2002.

Week of April 22, 2002—Tentative

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

Week of April 29, 2002—Tentative

Tuesday, April 30, 2002

9:30 a.m. Discussion of Intergovernmental Issues (Closed)

Wednesday, May 1, 2002

8:55 a.m. Affirmation Session (Public Meeting) (If needed)

9:00 a.m. Briefing on Results of Agency Action Review Meeting—Reactors (Public Meeting) (Contact: Robert Pascarelli, 301–415–1245)

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

Week of May 6, 2002—Tentative

There are no meetings scheduled for the Week of May 6, 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.

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: March 28, 2002.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 02–8035 Filed 3–29–02; 11:30 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from March 8, 2002 through March 21, 2002. The last biweekly notice was published on March 19, 2002 (67 FR 12597).

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 NRC'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 May 2, 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, which is available at the NRC'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 Systems (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. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: November 16, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Section 3.6.2.2, ''Suppression Pool Water Level,'' and TS 3.6.2.4, ''Suppression Pool Makeup (SMPU) System'' to revise the allowable operating range for the Suppression Pool water level and the modes of applicability for the upper containment pools. The amendment would permit draining of the reactor cavity pool portion of the upper containment pool with unit in Mode 3, ''Hot Shutdown,'' and reactor pressure less than 235 pounds per square inch gauge (psig). Draining of the upper containment pool is required as part of the refueling preparations and is currently not permissible in Mode 1, ''Power Operations,'' Mode 2, ''Startup,'' or Mode 3 by TS Section 3.6.2.4.

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

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

The proposed changes revise the required water levels in the upper containment pools and suppression pool during Mode 3. The probability of an accident previously evaluated is unrelated to the water levels in the pools since they are mitigative systems. The operation or failure of a mitigative system does not contribute to the occurrence of an accident. No active or passive failure mechanisms that could lead to an accident are affected by these proposed changes.

The consequences of a previously evaluated accident are not significantly increased. The changes have no impact on the ability of any of the Emergency Core Cooling Systems (ECCS) to function adequately, since adequate net positive suction head (NPSH) is provided with reduced water volumes. The post-accident containment temperature is not significantly affected by the proposed reduction in total heat sink volume. The increase in suppression pool water level to compensate for the reduction in upper containment pool volume will provide reasonable assurance that the minimum post-accident vent coverage is adequate to assure the pressure suppression function of the suppression pool is accomplished. The suppression pool water will be raised only after the reactor pressure has been reduced sufficiently to assure that the hydrodynamic loads from a loss of coolant accident will not exceed the design values. The reduced reactor pressure will also ensure that the loads due to main steam safety relief valve actuation with an elevated pool level are within the design loads. The change in exposure rate expected due to draining the upper containment pool in Mode 3 is small (i.e., by approximately two orders of magnitude) compared to the measured exposure rates in the reactor cavity during refueling preparations. Therefore, these changes do not have an adverse impact on the ability to maintain refueling exposure rates as low as reasonably achievable.

Therefore, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

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

The proposed changes to the water level requirements for the upper containment pool and the suppression pool do not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. The increased suppression pool water level does not increase the probability of flooding in the drywell. No new failures have been created by the change in the water level requirements.

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

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

The proposed changes to the upper containment pool and suppression pool water levels do not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the containment pressure suppression system is unchanged. The proposed total water volume is sufficient to provide high confidence that the pressure suppression and containment systems will be capable of mitigating large and small break accidents. All analyzed transient results remain well within the design values for the structures and equipment. 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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment requests: March 1, 2002.

Description of amendment requests: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714).

The licensee affirmed the applicability of the following NSHC determination in its request for amendments dated March 1, 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 proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

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

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

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

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

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: January 31, 2002.

Description of amendment request: Entergy Operations, Inc. is proposing that the Grand Gulf Nuclear Station, Unit 1, Operating License be amended to reflect a 1.7 percent increase in the licensed 100 percent reactor core thermal power level (an increase in reactor power level from 3,833 megawatts thermal to 3,898 megawatts thermal). These changes result from increased accuracy of the feedwater flow and temperature measurements to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation. The basis for this change is consistent with the revision, issued in June 2000, to appendix K to part 50 of title 10 of the *Code of Federal Regulations,* allowing operating reactor licensees to use an uncertainty factor of less than 2 percent of rated reactor thermal power in analyses of postulated design basis loss-of-coolant accidents.

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

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

Response: No.

The comprehensive analytical efforts performed to support the proposed change included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by this change. All systems and components will function as designed, and the applicable performance requirements have been evaluated and found to be acceptable.

The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of all components and systems that could be affected by this change. Evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. All systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable. Because the integrity of the plant will not be affected by operation at the uprated condition, it is concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the power calorimetric measurement allows the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 3,898 megawatts thermal (MWt). As such, all Final Safety Analysis Report (FSAR) Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to either bound operation at the 1.7 percent uprated condition, or new analyses were performed to verify all acceptance criteria continue to be met.

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

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

Response: No.

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

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

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

Response: No.

Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: January 31, 2002.

Description of amendment request: Entergy Operations, Inc. requests an amendment for the Grand Gulf Nuclear Station, Unit 1, Technical Specifications to extend the allowed out-of-service time from 72 hours to 14 days for a Division 1 or Division 2 Emergency Diesel Generator (DG) during reactor operational modes 1, 2, or 3. The proposed changes are intended to provide flexibility in performance of corrective and preventive maintenance on the DGs during power operation.

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

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

Response: No.

The proposed Technical Specification (TS) changes do not affect the design, operational characteristics, function, or reliability of the DGs. The DGs are not the initiators of previously evaluated accidents. The DGs are designed to mitigate the consequences of previously evaluated accidents including a loss of offsite power. Extending the allowed outage time (AOT) for a single DG would not significantly affect the previously evaluated accidents since the remaining DGs supporting the redundant ESF systems would continue to perform the accident mitigating functions as designed.

The duration of a TS AOT is determined considering that there is a minimal possibility that an accident will occur while a component is removed from service. A risk-informed assessment was performed which concluded that the increase in plant risk is small and consistent with the USNRC [U.S. Nuclear Regulatory Commission] ''Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,'' **Federal Register**, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986, as further described by NRC [Nuclear Regulatory Commission] Regulatory Guide 1.177.

The current TS requirements establish controls to ensure that redundant systems relying on the remaining DGs are Operable. In addition to these requirements, administrative controls will be established to provide assurance that the AOT extension is not applied during adverse weather conditions that could potentially affect offsite power availability. Administrative controls are also implemented to avoid or minimize risk-significant plant configurations during the time when a DG is removed from service.

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

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

Response: No.

The proposed TS changes do not involve a change in the design, configuration, or method of operation of the plant that could create the possibility of a new or different kind of accident. The proposed change extends the AOT currently allowed by the TS.

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

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

Response: No.

The Engineered Safety Feature (ESF) systems required to mitigate the consequences of postulated accidents consist of three independent divisions. The ESF systems of any two of the three divisions provide for the minimum safety functions necessary to shut down the unit and maintain it in a safe shutdown condition. Each of the three independent ESF divisions can be powered from one of the offsite power sources or its associated on-site DG. This design provides adequate defense-in-depth to ensure that the ESF equipment needed to mitigate the consequences of an accident will have diverse power sources available to accomplish the required safety functions. Thus, with one DG out of service, there are sufficient means to accomplish the safety functions and prevent the release of radioactive material in the event of an accident.

The proposed AOT change does not affect any of the assumptions or inputs to the safety analyses of the FSAR and does not erode the decrease in severe accident risk achieved with the issuance of the Station Blackout (SBO) Rule, 10 CFR 50.63 ''Loss of All Alternating Current Power.''

The proposed extended AOT deviates from the recommended 72 hour AOT of Regulatory Guide (RG) 1.93. However, an extension of the 72 hour AOT to 14 days has been demonstrated to be acceptable based on deterministic and risk-informed analyses. The proposed changes are not in conflict with any other approved codes or standards applicable to the onsite AC [Alternating Current] power sources.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 8, 2002.

Description of amendment request: The proposed amendment would replace referenced control requirements for access to high radiation areas with the actual requirements of 10 CFR part 20. The referenced document in Technical Specifications Section 6.11 would no longer exist.

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?

The proposed changes replace referenced control requirements affecting access to high radiation areas with the actual requirements. This proposed change does not involve any changes to system or equipment configuration. The reliability of systems and components relied upon to prevent or mitigate the consequences of accidents previously evaluated is not affected by the proposed changes. 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?

The proposed changes are administrative in nature and do not involve a change to the plant design or operation. No new or different types of equipment will be installed as a result of this change. The proposed change is administrative in nature and replaces referenced control requirements for access to high radiation areas with the actual requirements. No new accident modes or equipment failure modes are created by these changes. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change does not impact or have a direct effect on any safety analysis assumptions. The proposed change is administrative in nature and replaces referenced control requirements for access to high radiation areas with the actual requirements.

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 & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Robert A. Gramm.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of amendment requests: January 14, 2002.

Description of amendment requests: The proposed amendments would add an allowable plus or minus (±) 1 percent (%) as-left setpoint tolerance for the pressurizer code safety valves to Unit 1 and Unit 2 technical specification (TS) 3.4.2 and TS 3.4.3. In addition, the proposed amendments would revise Unit 2 TS 3.4.2 and TS 3.4.3 to increase the allowable as-found setpoint tolerance for the Unit 2 pressurizer code safety valves from ± 1 % to ± 3%.

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

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

Response: No.

Probability of Occurrence of an Accident Previously Evaluated—

The proposed changes to pressurizer code safety valve as-found and as-left setpoint tolerance do not affect any accident initiators or precursors. There are no new failure modes for the pressurizer code safety valves created by this change in setpoint tolerance. No adverse interactions with the RCS are created by this change in setpoint tolerance. The lowest possible setpoint of any of the pressurizer code safety valves (including the ± 3% tolerance) is higher than the highest RCS pressures anticipated during shutdown, startup, normal operating, and anticipated operational occurrence conditions. The lowest possible pressurizer code safety valve setpoint is also higher than the setpoint of the PORVs. Therefore, there would not be an adverse interaction between the pressurizer code safety valves and the PORVs. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

The format changes for the Unit 2 TS 3.4.3 page do not impact any accident initiators or precursors. Thus, the probability of occurrence of an accident previously evaluated is not significantly increased.

Consequences of an Accident Previously Evaluated—

The proposed change to add an allowable as-left setpoint tolerance for the Unit 1 and 2 pressurizer code safety valves does not adversely affect any of the accident and safety analyses. In addition, the proposed increase in the Unit 2 as-found pressurizer code safety valve setpoint tolerance does not adversely affect any of the accident and safety analyses. Both the as-left setpoint of ± 1% and the as-found setpoint of ± 3% of the nominal lift pressure of 2485 psig provides reasonable assurance that the pressurizer code safety valves are capable of performing their design function as assumed in the accident and safety analyses. Even at the highest allowable lift pressure, the pressurizer code safety valves, in conjunction with the RPS, remain capable of limiting the RCS pressure within the Safety Limit of 110% of design pressure (or 2735 psig). Thus, there will be no increase in offsite doses and the consequences of an accident previously analyzed are not increased.

The format changes for the Unit 2 TS 3.4.3 page do not impact the pressurizer code safety valve's function. Thus, there will be no increase in offsite doses, and the consequences of an accident previously analyzed are not increased.

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

Response: No.

The proposed changes to pressurizer code safety valve as-found and as-left setpoint tolerance do not create any new or different accident initiators or precursors. There are no new failure modes for the pressurizer code safety valves created by this change in setpoint tolerance. No adverse interactions with the RCS are created by this change in setpoint tolerance. The lowest possible setpoint of any of the pressurizer code safety valves (including the ± 3% tolerance) is higher than the highest RCS pressures anticipated during shutdown, startup, normal operating, and anticipated operational occurrence conditions. The lowest possible pressurizer code safety valve setpoint is also higher than the setpoint of the PORVs. Therefore, there would not be an adverse interaction between the pressurizer code safety valves and the PORVs. Thus, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The format changes for the Unit 2 TS 3.4.3 page do not create any new or different accident initiators or precursors. Thus, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

Response: No.

The proposed changes do not impact pressurizer code safety valve capability to perform the design function required by the accident and safety analyses, nor do the proposed changes impact the operational characteristics of the pressurizer code safety valves. The pressurizer code safety valves, in conjunction with the RPS, ensure that the RCS Safety Limit of 110% of design pressure (or 2735 psig) is not exceeded for any analyzed event. Therefore, the proposed changes do not involve a significant reduction in margin of safety.

The format changes for the Unit 2 TS 3.4.3 page 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 requests involve no significant hazards consideration.

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

NRC Section Chief: William D. Reckley, Acting Section Chief.

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

Date of amendment request: January 21, 2002.

Description of amendment request: The proposed revised Technical Specification (TS) Requirement will modify TS Surveillance Requirement (SR) 3.7.3.1 to improve consistency with Cooper Nuclear Station (CNS) License Amendment No. 185, approved on March 13, 2001, and eliminate unnecessary restrictions regarding how the Reactor Equipment Cooling (REC) System surge tank level is monitored.

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?

This change eliminates the specific details regarding performing the SR 3.7.3.1 verification of Reactor Equipment Cooling (REC) surge tank level. This change will not result in a significant increase in the probability of an accident previously evaluated because the method of verifications of REC surge tank level has no effect on the initiators of any analyzed events.

The method of performing the surveillance on REC surge tank level does not affect the performance of the minimum equipment credited in the mitigation of any analyzed event. As a result, no analysis assumptions or mitigative functions are impacted. Therefore, this change will not result in a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There is no change being made to the parameters within which the plant is operated. There are no setpoints, at which protective or mitigative actions are initiated, affected by this change. This change will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alteration in the procedures which ensure the plant remains within analyzed limits is being proposed, and no change is being made to the procedures relied upon to an off-normal event. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. Credited equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed change is acceptable because the operability of the REC System is unaffected, there is no detrimental impact on any equipment design parameter, and the plant will still be required to operate within assumed conditions. The normal procedural controls on methods of surveillance performance provide adequate assurance that the REC System will be capable of performing its intended safety function. Detailing the performance method within the TSs does not impact the margin of safety (which is more closely related to tank volume than the method of verifying volume). Therefore, the 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. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499.

NRC Section Chief: Robert A. Gramm.

Nuclear Management Company, LLC, Docket No. 50–331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: February 8, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to change TS Section 5.0, Administrative Controls, to adopt TSTF–258 Revision 4. Revisions to the TS are proposed to Section 5.2.2, Unit Staff, to delete details of staffing requirements and delete requirements for the Shift Technical Advisor (STA) as a separate position while retaining the function. Section 5.5.4, Radioactive Effluent Controls Program, would be revised to be consistent with the intent of 10 CFR part 20. Section 5.6.4, Monthly Operating Reports, would be revised by deleting periodic reporting requirements for main steam safety/relief valve challenges to be consistent with Generic Letter 97–02. Section 5.7, High Radiation Area, would be revised in accordance with 10 CFR 20.1601(c).

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

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

This request for amendment to Duane Arnold Energy Center's TS provides for adoption of the NRC-approved generic change TSTF item TSTF–258, Revision 4. The Amendment request includes revisions to TS Section 5.0, ''Administrative Controls,'' to delete details of staffing requirements, delete requirements for the STA as a separate position while retaining the function, revise the Radioactive Effluent Controls Program to be consistent with the intent of 10 CFR 20, delete periodic reporting requirements of challenges to main steam safety/relief valves, and revise radiological control requirements for radiation areas to be consistent with those specified in 10 CFR 20.1601(c).

The proposed TS changes are administrative in nature and do not impact 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 mitigation of accidents or transient events. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed TS changes are administrative in nature and do not alter plant configuration, require that new equipment be installed, alter assumptions made about accidents previously evaluated or impact the operation or function of plant equipment or systems. The proposed changes do not introduce any new modes of plant operation or make any changes to system setpoints. The proposed changes do not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated has not been created.

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

The proposed TS changes are administrative in nature and do not involve physical changes to plant structures, systems, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed changes do not involve a change to any safety limits, limiting safety system settings, limiting conditions for operation, or design parameters for any SSC. The proposed changes do not impact any safety analysis assumptions and do not involve a change in initial conditions, system response times, or other parameters affecting any accident analysis. Regarding the deletion of the requirement for the STA as a separate position, the function will be retained, so there will be no reduction in the margin of safety. As a result, there is no 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. Alvin Gutterman, Morgan Lewis, 1111 Pennsylvania Avenue NW., Washington, DC 20004.

NRC Section Chief: William D. Reckley, Acting Section Chief.

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

Date of amendment request: February 12, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement (SR) 4.0.E to extend the delay period before entering a limiting condition for operation following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours or up to the limit of the time interval, whichever is less'' to ''* * * up to 24 hours or up to the limit of the time interval, whichever is greater.'' In addition, the following requirement would be added to SR 4.0.E: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001 (66 FR 32400), on possible amendments concerning missed surveillances, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line-item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on September 28, 2001 (66 FR 49714). The licensee affirmed the applicability of the following NSHC determination in its application dated February 12, 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 proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

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

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

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

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

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

NRC Section Chief: William D. Reckley, Acting.

Portland General Electric Company, et al., Docket No. 50–344, Trojan Nuclear Plant, Columbia County, Oregon

Date of amendment request: November 15, 2001, as supplemented by letter dated January 31, 2002.

Description of amendment request: The proposed amendment request modifies License Condition 2.C(10) associated with loading and contingency unloading of spent fuel casks in the fuel building due to changes in the dry storage system design.

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

Accidents previously evaluated are those addressed in the Trojan Nuclear Plant (TNP) Defueled Safety Analysis Report (DSAR), the TNP Decommissioning Plan and License Termination Plan (''Decommissioning Plan''), and LCA [license change application] 237, Revision 3, and LCA 246, Revision 0. [Since their approval via Amendments 199 and 200 to the TNP License on April 23, 1999, Revision 3 of LCA 237 and Revision 0 of LCA 246, have undergone revision per 10 CFR 50.59, as allowed by TNP License Condition 2.C(10). The current revisions are LCA 237, Revision 4, and LCA 246, Revision 1.] The basis for the conclusion that the probability or consequences of an accident previously evaluated in the DSAR or Decommissioning Plan is not significantly increased is not materially changed from the significant hazards consideration determination provided in the current LCA 237, Revision 4, and LCA 246, Revision 1. Loading and contingency unloading of the MPC [multi-purpose canister] as described in the proposed Revision 5 of LCA 237 and Revision 2 of LCA 246 consist of activities that are functionally the same as those for loading and contingency unloading a PWR [pressurized water reactor] Basket under the previous Trojan Storage System design. With the original Transfer Cask, PWR Basket, and its shield and structural lids and associated welds replaced under the new design by the Holtec Transfer Cask, MPC, and its MPC redundant closures (i.e., lid, vent and drain port cover plates, closure ring, and associated welds), respectively, these and associated Trojan Storage System design changes do not significantly impact the activities that will be conducted during ISFSI [independent spent fuel storage installation] loading/unloading. Furthermore, the safety evaluations in the proposed Revision 5 of LCA 237 and Revision 2 of LCA 246 show that the Trojan ISFSI design changes do not significantly impact the potential for or consequences of off-normal events or accidents during ISFSI loading and contingency unloading. Thus, the probability or consequences of an accident previously evaluated in the DSAR or Decommissioning Plan is not significantly increased.

The postulated events previously evaluated in Revision 3 of LCA 237 and Revision 0 of LCA 246 include drops, tipovers, mishandling, operational errors, and support system malfunctions that could potentially occur during loading and contingency unloading operations.

As discussed in proposed Revision 5 to LCA 237 and Revision 2 to LCA 246, the Trojan Storage System design changes do not significantly affect the conclusions with respect to the potential for or consequences of a Transfer Cask and/or MPC drop, tipover, or mishandling event. The design safety factors, load testing requirements, and administrative controls (i.e., procedures, training, maintenance, and inspections) for the fuel handling equipment are materially unaffected by the Trojan Storage System design changes, such that there is no significant increase in probability of a Transfer Cask and/or MPC drop, tipover, or mishandling event. As described in the safety evaluation in proposed Revision 5 to LCA 237 and Revision 2 to LCA 246, the calculated consequence of a Transfer Cask drop, tipover, or mishandling event prior to the MPC lid being welded to the MPC is approximately 0.003 rem whole body dose at the site boundary, which is the same as was calculated for these events in LCA 237, Revision 3. This calculated consequence, which is well below the EPA PAG [Environmental Protection Agency protective action guide] of 1 rem whole body dose for the early phase of an event, has accumulated additional conservatism since the submittal and NRC approval of LCA 237, Revision 3, applicable to loading the PWR Basket. The additional conservatism is the result of the calculation assumption that five years have elapsed for cooling of the fuel, combined with the fact that approximately five additional years have passed since this event was originally analyzed for LCA 237, Revision 3, during which additional cooling of the TNP spent nuclear fuel has occurred. Thus, there is no significant increase in consequences of a Transfer Cask drop, tipover, or mishandling event.

The Trojan Storage System design changes also do not significantly increase the probability or consequences of operational errors and/or support system malfunctions that could potentially occur during loading/unloading operations. As discussed in the safety evaluation in proposed Revision 5 to LCA 237 and Revision 2 to LCA 246, the changes to pressures associated with the ISFSI confinement boundary do not impact the conclusion that the postulated inadvertent over-pressurization of the MPC during draining and/or drying operations is not considered credible, since multiple equipment failures and a procedural error are still required in order for the event to occur. With the revised design decay heat load as summarized above, the longer time period required for boiling to occur in the MPC further reduces the potential for a postulated over-pressurization event.

As shown in proposed Revision 5 of LCA 237 and Revision 2 of LCA 246, the higher operating pressures during loading operations (*e.g.,* pressure testing and MPC blowdown and backfill pressures) and the redesign of several of the systems involved in MPC closure operations (*e.g.,* vacuum drying, blowdown system, and helium recirculation cooling), do not significantly impact the probability or consequences of equipment failures. The maximum normal design pressure ratings of the MPC, vacuum drying system, helium recirculation system, and helium backfill system, including their associated pressurized lines and system components, are such that the operating pressure increase does not significantly increase the probability of a passive failure of a pressurized line on the MPC. However, because of the increased operating and test pressures associated with the Holtec-designed MPC as compared to the PWR Basket, the consequence of a bounding scenario involving the passive failure of a pressurized line is increased. However, this increase is not considered to be significant since, as detailed in Section 5.2.5.2.2 of proposed Revision 5 to LCA 237 and Revision 2 to LCA 246, the dose consequence remains well below the EPA PAG of 1 rem whole body for the early phase of an event.

Based on the above, the impacts of the Trojan Storage System design changes on cask loading/unloading operations would not significantly increase the probability or consequences of any accident previously evaluated.

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

The aforementioned design changes for the Trojan Storage System do not create the possibility of a new or different kind of accident from any accident previously evaluated, including those evaluated in Revision 3 of LCA 237 and Revision 0 of LCA 246 approved by the NRC on April 23, 1999. With the original Transfer Cask, PWR Basket, and its shield and structural lids and associated welds replaced under the new design by the Holtec Transfer Cask, MPC, and its MPC redundant closures (i.e., lid, vent and drain port cover plates, closure ring, and associated welds), respectively, these and associated Trojan Storage System design changes do not significantly impact the functional activities that will be conducted during ISFSI loading/unloading. Thus, the loading procedure and system design changes do not introduce any new types of accidents not previously analyzed in Revision 3 of LCA 237 and Revision 0 of LCA 246.

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

The basis for the conclusion that a significant reduction in the margin of safety is not involved is not materially changed from the significant hazards consideration determination provided in the current LCA 237, Revision 4, and LCA 246, Revision 1. Specifically, the TNP Permanently Defueled Technical Specifications (PDTS) contain four limiting conditions of operation that address: (1) Spent Fuel Pool water level, (2) Spent Fuel Pool boron concentration, (3) Spent Fuel Pool temperature, and (4) Spent Fuel Pool load restrictions. These Technical Specifications will remain in effect as long as spent fuel is stored in the Spent Fuel Pool, which is in accordance with their applicability statements. As discussed below, the Trojan Storage System design changes and their impact on ISFSI loading/unloading activities will not affect the PDTS or their bases.

Loading and contingency unloading of the MPC as described in the proposed Revision 5 of LCA 237 and Revision 2 of LCA 246 consist of activities that are functionally the same as those for loading and contingency unloading a PWR Basket under the previous Trojan Storage System design. The Cask Loading Pit, where spent fuel will be loaded into the MPC, is immediately adjacent to the Spent Fuel Pool. The gate between the Cask Loading Pit and Spent Fuel Pool will be opened to allow spent fuel assemblies to be moved from the spent fuel storage racks in the Spent Fuel Pool to the MPC in the Cask Loading Pit. Opening the gate will allow free exchange of the water between the Cask Loading Pit and the Spent Fuel Pool. The water in the Cask Loading Pit must be at essentially the same level, boron concentration, and temperature as the Spent Fuel Pool prior to the first opening of the gate to ensure that the limiting conditions of operation are continuously satisfied for the Spent Fuel Pool. Therefore, the Cask Loading Pit will be filled, to about the same level as the Spent Fuel Pool, with water that is about the same boron concentration and temperature as the Spent Fuel Pool. With these precautions, the limiting conditions of operation pertaining to Spent Fuel Pool level, boron concentration, and temperature will be continuously maintained for the Spent Fuel Pool and the margin of safety will be unaffected. Except for small changes to accommodate lid lift rigging, the level in the Cask Loading Pit will not be reduced until the MPC lid has been placed on the loaded MPC. This configuration is consistent with the objective of keeping the radiological exposure to personnel as low as reasonably achievable (ALARA). The contingency unloading sequence is essentially the reverse of the loading sequence. Thus, the loading and contingency unloading processes for the MPC with the Trojan Storage System design changes incorporated do not involve a significant reduction in the margin of safety.

As with the previous design, the Trojan Storage System design changes will be implemented such that when lifting and moving heavy loads, loads that will be carried over fuel in the Spent Fuel Pool racks and the heights at which they may be carried will be limited in such a way as to preclude impact energies, in the unlikely event of a drop, from exceeding 240,000 in-lbs in accordance with Limiting Condition for Operation (LCO) 3.1.4, ‘‘Spent Fuel Pool Load Restrictions.’’ With this precaution, the LCO pertaining to load restrictions over the Spent Fuel Pool will be satisfied for fuel stored in the Spent Fuel Pool racks and the margin of safety will be unaffected. The safe load path for heavy loads being lifted and moved outside the Spent Fuel Pool will be located sufficiently far from the Spent Fuel Pool as to not have an adverse effect on the Spent Fuel Pool in the unlikely event of a load drop. In addition, the Trojan Storage System design changes do not affect the implementation of mechanical stops and electrical interlocks on the Fuel Building overhead crane that provide additional assurance that heavy loads are not carried over the fuel in the Spent Fuel Pool racks. Thus, the Trojan Storage System design changes and their impact on ISFSI loading and contingency unloading activities do 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: Douglas R. Nichols, Esq., Portland General Electric Company, 121 S.W. Salmon Street, Portland, Oregon 97204.

NRC Section Chief: Robert A. Gramm.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendment request: October 30, 2001, as supplemented by letter dated February 11, 2002.

Description of amendment request: The proposed amendments would revise Technical Specifications Table 3.3.1–1, ''Reactor Trip System Instrumentation'' and the associated Bases B 3.3.1. A limit or ''clamp'' on the Over Temperature Delta Temperature (OTDT) reactor trip function is proposed to address design issues related to fuel rod design under transient conditions. In addition, editorial revisions to Bases B 3.3.1 are included.

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

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

The proposed clamp on the OTDT reactor trip function is not credited in the safety analyses. Implementation of the limit or ''clamp'' on the OTDT reactor trip function, along with the corresponding changes to the AFD [axial flux difference] modifier f_1 (AFD) and RAOC [relaxed axial offset control] band, will ensure the prevention of stress failure of the fuel rod cladding for Condition I and II reactor coolant system cooldown events. This demonstrates continued compliance with 10 CFR 50, Appendix A, Criterion 10, *i.e.*, that the specified acceptable fuel design limits are not exceeded.

There is no change in the radiological consequences of any accident since the fuel clad, the reactor coolant system pressure boundary, and the containment are not changed, nor will the integrity of these physical barriers be challenged. In addition, the proposed modification will not change, degrade, or prevent any reactor trip system actuations.

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

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

The proposed clamp on the OTDT reactor trip function is not credited in the safety analyses. Implementation of the limit or ''clamp'' on the OTDT reactor trip function, along with the corresponding changes to the AFD modifier f_1 (AFD) and RAOC band, will ensure the prevention of stress failure of the fuel rod cladding for Condition I and II reactor coolant system cooldown events.

The design basis of the OTDT reactor trip setpoint is to ensure DNB [departure from nucleate boiling] protection and to preclude vessel exit boiling. The installation of the OTDT clamp would continue to ensure this same protection and that the OTDT design basis would remain unaffected. The introduction of the OTDT clamp would not create any new transients nor would it invalidate the OTDT design basis. In addition, there are no transients analyzed in the VEGP [Vogtle Electric Generating Plant] FSAR [final safety analysis report] that result in a reduction in the reactor coolant temperature which rely on OTDT as the primary reactor trip function, as cooldown events tend to be non-limiting with respect to the criterion of DNB.

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?

The proposed clamp on the OTDT reactor trip function is not credited in the safety analyses. Implementation of the limit or ''clamp'' on the OTDT reactor trip function, along with the corresponding changes to the AFD modifier f_1 (AFD) and RAOC band, will ensure the prevention of stress failure of the fuel rod cladding for Condition I and II RCS [reactor coolant system] cooldown events. This demonstrates continued compliance with 10 CFR 50, Appendix A, Criterion 10, i.e., that the specified acceptable fuel design limits are not exceeded.

The design basis of the OTDT reactor trip setpoint is to ensure DNB [departure from nucleate boiling] protection and to preclude vessel exit boiling. The installation of the OTDT clamp would continue to ensure this same protection and that the OTDT design basis would remain unaffected.

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. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard J. Laufer, Acting.

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: February 14, 2002.

Description of amendment request: The proposed amendment revises Technical Specifications (TS) 3.4.2.2, ''Reactor Coolant System.'' to relax the lift setting tolerance of the pressurizer safety valves from ±2 percent to ±3 percent. The current TS requirements that the as left lift setting be within ±1 percent will remain intact.

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

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

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR [Updated Final Safety Analysis Report]. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed TS change takes credit for the assumptions made in the reanalysis of the turbine trip and rod withdrawal from power events already evaluated in the UFSAR. Those analyses demonstrated that (1) the fuel design limits were maintained by the reactor protection system since the DNBR [departure from nucleate boiling ratio] was maintained above the limit value, and (2) the plant design is such that a turbine trip presents no hazard to the integrity of the RCS [reactor coolant system] or the main steam system pressure boundary. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 14, 2002.

Description of amendment request: The proposed amendment revises Technical Specifications to eliminate shutdown actions associated with radiation monitoring instrumentation. The proposed changes will enhance plant reliability by reducing exposure to unnecessary shutdowns and increase operational flexibility, and relax certain other restrictions.

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

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

Response: No.

The radiation monitors affected by the proposed amendment are not potential accident initiators. Adequate measures are available to compensate for radiation monitors that are out of service. The proposed amendment does not affect how the affected radiation monitors function or their role in the response of an operator to an accident or transient. The core damage frequency in the STP [South Texas Project] PRA [probabilistic risk assessment] is not impacted by the proposed changes. Therefore, STPNOC [South Texas Project Nuclear Operating Company] concludes that there is no significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The radiation monitors affected by the proposed amendment are not credited for the prevention of any accident not evaluated in the safety analysis. The proposed amendment involves no changes in the way the plant is operated or controlled. It involves no change in the design configuration of the plant. No new operating environments are created. Therefore, STPNOC concludes the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change has no significant effect on functions that are supported by the affected radiation monitors. There will be no significant effect on the availability and reliability of the affected radiation monitors. Adequate measures are available to compensate for radiation monitors that are out of service. Therefore, STPNOC concludes 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 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: Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: February 14, 2002.

Description of amendment request: The proposed amendment revises Technical Specifications governing radiation monitoring instrumentation and reactor coolant system leakage detection to eliminate the associated shutdown action requirements and relax certain other restrictions.

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

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

Response: No.

The radiation monitors and leakage detection instrumentation affected by the proposed amendment are not potential accident initiators. Adequate measures are available to compensate for instrumentation that is out of service. The proposed amendment does not affect how the affected instrumentation normally functions or its role in the response of an operator to an accident or transient. The core damage frequency in the STP [South Texas Project] PRA [probabilistic risk assessment] is not impacted by the proposed changes. Therefore, STPNOC [South Texas Project Nuclear Operating Company] concludes that there is no significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The instrumentation affected by the proposed amendment is not credited for the prevention of any accident not evaluated in the safety analysis. The proposed amendment involves no changes in the way the plant is operated or controlled. It involves no change in the design configuration of the plant. No new operating environments are created. Therefore, STPNOC concludes the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change has no significant effect on functions that are supported by the affected instrumentation. There will be no significant effect on the availability and reliability of the affected instrumentation. Adequate measures are available to compensate for instrumentation that is out of service. Therefore, STPNOC concludes 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 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: Morgan Lewis, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

Tennessee Valley Authority, Docket No. 50–390, Watts Bar Nuclear Plant, Unit 1, Rhea County, Tennessee

Date of amendment request: January 14, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.4.16, applicable Bases ''Reactor Coolant System Specific Activity,'' and Surveillance Requirement (SR) 3.4.16.2, from 1.0 microcuries per gram (uCi/gm) iodine-131 to 0.265 uCi/gm iodine-131. TS 3.4.16, Figure 3.4.16–1, ''Reactor Coolant Dose Equivalent Iodine-131 Specific Activity Limit Versus Percent of Rated Thermal Power,'' is being deleted and the maximum value of 21 uCi/gm iodine-131 is being added to TS Required Action 3.14.16.A and 3.4.16.C. In addition, TS Section 3.3.7, ''CREVS [Control Room Emergency Ventilation System] Actuation Instrumentation,'' Table 3.3.7–1 changes the allowable value to the Control Room Radiation and Control Room Air Intakes for SR 3.3.7.1, 3.3.7.2, and 3.3.7.4 from less than or equal to (≤) 5.77E–04 uCi/cubic centimeter (cc) (20,199 counts per minute (cpm)) to ≤9.45E–05 uCI/cc (3,307 cpm).

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

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

The proposed Technical Specification[s] change[s] to reduce the steady state and 48[-]hour reactor coolant system (RCS) allowable iodine concentrations, and to revise the surveillance requirement value for the Main Control Room [MCR] air intake radiation monitors [do] not change any operator actions nor [do they] change plant systems or structures. Therefore, the proposed change[s] to WBN Unit 1 Technical Specification[s] [do] not result in a significant increase in the probability of an accident.

The calculated radiological consequences at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are larger than currently discussed in the Final Safety Analysis Report (FSAR) accidents for the main steam line break (MSLB) and steam generator tube rupture (SGTR) (with the exception of thyroid and beta doses being slightly lower for STGR) accidents. The radiological consequences for the SGTR and MSLB accidents increased due to utilizing more conservative methodologies and more conservative assumptions in the calculation. However, the calculated radiological consequences remain within the limits identified in 10 CFR 100, ''Reactor Site Criteria,'' and General Design Criteria (GDC)–19, ''Control Room,'' and are consistent with NUREG–0800, ''Standard Review Plan,'' acceptance criteria.

The surveillance requirement radiation limit for the Main Control Room air intake radiation monitors will be reduced to compensate for the change in source terms which resulted from the use of the methodology changes in the SGTR accident. This change ensures the monitors perform their safety function of control room isolation during accident conditions and does not increase the probability or consequences of an accident previously evaluated.

In summary, the control room dose, the LPZ dose, and the EAB dose for the SGTR and MSLB remain bounded by the acceptance criteria of NUREG–0800 and continue to satisfy an appropriate fraction of the 10 CFR 100 dose limits and the GDC–19 dose limits. The surveillance requirement changes for the Main Control Room radiation monitors ensure the monitors perform their intended design function. Therefore, the proposed change does not result in a significant increase in the [probability or] consequences of an accident previously analyzed.

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

The proposed TS change does not alter the configuration of the plant. The changes do not directly affect plant operation. The change will not result in the installation of any new equipment or system or the modification of any existing equipment or systems. No new operation procedures, conditions or modes will be created by this proposed change. Therefore, 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 methods for calculating the radiological consequences are revised for the MSLB and SGTR analysis to utilize the thyroid dose conversion factors in International Commission on Radiation Protection Publication 30 (ICRP–30) to calculate the dose and ARCON96 methodology to calculate atmospheric dispersion coefficients.

The calculated radiological consequences at the EAB and LPZ are slightly larger than those noted in the FSAR accidents for the MSLB and SGTR (thyroid and beta doses slightly lower for SGTR) accidents. The radiological dose consequences for the SGTR and MSLB accidents increased due to utilizing more conservative methodologies and more conservative assumptions in the calculation. The calculated dose consequences of the evaluated accidents remain less than the dose limits identified in 10 CFR 100 and GDC–19, and are consistent with NUREG–0800 acceptance criteria. The surveillance requirement for the MCR radiation monitors is being reduced for consistency with lower source terms and to ensure the monitors perform their intended design function of isolating the Main Control Room subsequent to an accident. Therefore, it is concluded that the proposed change to lower the RCS Specific Activity and subsequent changes to the Main Control Room radiation monitors are required to ensure the Main Control Room dose and the offsite dose are below the acceptable limits. Therefore these changes do not result in a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/reading-rm.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: August 13, 2001.

Brief description of amendment: The amendment defers withdrawal of the first set of reactor vessel surveillance specimens until 10.4 effective full power years, expected to be one additional operating cycle.

Date of issuance: March 8, 2002.

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

Amendment No.: 143.

Facility Operating License No. NPF–62: The amendment changes the updated safety analysis report.

*Date of initial notice in **Federal Register:*** October 17, 2001 (66 FR 52796). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 21, 2001, as supplemented by letter dated January 18, 2002.

Brief description of amendment: The amendment modifies the technical specification requirement that the main steamline safety relief valves (SRVs) open when they are manually actuated by instead requiring that the SRV valve actuators stroke on a manual actuation.

Date of issuance: March 19, 2002.

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

Amendment No.: 144.

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

*Date of initial notice in **Federal Register:*** October 3, 2001 (66 FR 50465). The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 19, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 9, 2001.

Brief description of amendments: The amendments would revise Technical Specification 5.6.5b to add NRC-approved Topical Report CENPD–404–P–A, ''Implementation of ZIRLO $_{TM}$ Cladding Material in CE Nuclear Power Fuel Assembly Designs,'' into the list of analytical methods used to determine core operating limits and thus, enable use of ZIRLO clad fuel in Palo Verde Nuclear Generating Station units.

Date of Issuance: March 12, 2002.

Effective date: March 12, 2002, and shall be implemented within 60 days of the date of issuance.

Amendment Nos.: Unit 1–140, Unit 2–140, Unit 3–140.

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

*Date of initial notice in **Federal Register:*** January 22, 2002 (67 FR 2919). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2002.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket Nos. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: November 26, 2001, as supplemented January 31, 2002, February 5, 2002, and February 11, 2002.

Description of amendment request: The amendment revises the Improved Technical Specification 5.5.12 to allow a one-time interval increase for the Type A Integrated Leakage Rate Test for no more than 3 years, 2 months.

Date of issuance: March 6, 2002.

Effective date: March 6, 2002.

Amendment Nos: 216.

Facility Operating License No. DPR–71: The amendment changes the Technical Specifications.

*Date of initial notice in **Federal Register:*** January 8, 2002 (67 FR 926). The January 31, 2002, and February 5, 2002, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice. The February 11, 2002, supplement revised the original request, but the initial no significant hazards consideration determination bounded the revised request.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 6, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: June 26, 2001, as supplemented January 14, and February 1, 2002.

Description of amendment request: The amendments revise the Technical Specifications to support installation of the General Electric Nuclear Measurement Analysis and Control Digital Power Range Neutron Monitoring System.

Date of issuance: March 8, 2002.

Effective date: March 8, 2002.

Amendment Nos: 217 and 243.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38759). The January 14, and February 1, 2002, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 1, 2001, as supplemented February 4, 2002.

Description of amendment request: The amendment revises the Technical Specifications to incorporate NRC-approved Technical Specification Task Force Traveler Item 51, ''Revise containment requirements during handling irradiated fuel and core alterations,'' Revision 2.

Date of issuance: March 14, 2002.

Effective date: March 14, 2002.

Amendment Nos: 218 and 244.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46477). The February 4, 2002, supplement contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 2002.

No significant hazards consideration comments received: No.

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

Date of amendment request: November 7, 2001.

Description of amendment request: The amendments revise Technical Specification (TS) 3.1.4, ''Control Rod Scram Times,'' to delineate more specific requirements for testing control rod scram times following refueling outages. TS 5.1 is revised to reference Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59. The amendment incorporates the Nuclear Regulatory Commission-approved Technical Specification Task Force (TSTF) Item 222, Revision 1, ''Control Rod Scram Testing,'' and TSTF Item 364, Revision 0, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59.''

Date of issuance: March 19, 2002.

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

Amendment Nos: 219/245.

Facility Operating License Nos. DPR–71 and DPR–62: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59502). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 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: August 6, 2001.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.3.2 for Engineered Safety Feature Actuation System Instrumentation, and TS 3.3.6 for Containment Purge and Exhaust Isolation Instrumentation. The amendments excluded the Containment Purge Ventilation System and the Hydrogen Purge System containment isolation valves from the instrumentation testing requirements in TS 3.3.2 and TS 3.3.6. The amendments also made appropriate changes in the Bases for TS 3.3.6 and TS 3.6.3.

Date of issuance: March 20, 2002.

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

Amendment Nos.: 196/189.

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

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64291). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 2002.

No significant hazards consideration comments received: No.

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

Date of application of amendments: August 14, 2001.

Brief description of amendments: The proposed amendments would revise TS Surveillance Requirement 3.3.5.2 by changing the Engineered Safeguards Protective System Analog Instrument channel functional test frequency from 31 days to 92 days.

Date of Issuance: March 18, 2002.

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

Amendment Nos.: 321/321/322.

Renewed Facility Operating License Nos. DPR–38, DPR–47, and DPR–55: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 5, 2001 (66 FR 46478). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 18, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: December 21, 2001, as supplemented February 15, 2002.

Brief description of amendment: This amendment revises the minimum critical power ratio safety limits for operating cycle 10.

Date of issuance: March 12, 2002.

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

Amendment No.: 156.

Facility Operating License No. NPF–39: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** January 22, 2002 (67 FR 2924). The February 15, 2002, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 23, 2001.

Brief description of amendments: These amendments deleted Technical Specification 3.4.2, Limiting Condition for Operation, Action Statement b, concerning operator actions with stuck open safety/relief valves.

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

Effective date: March 20, 2002.

Amendment Nos.: 157 and 119.

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44171). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 26, 2001, as supplemented by letter dated November 15, 2001.

Brief description of amendments: The amendments revised Technical Specification 3/4.3.3, Emergency Core Cooling System, Actions 36 and 37 of Table 3.3.3–1, and associated TS Bases. The change to Action 36 clarifies equipment affected by inoperable components. The change to Action 37 takes advantage of the inherent overlap of the degraded voltage relays' characteristics such that inoperable relays that define a channel can be taken out of service without placing its associated source breaker in the trip position.

Date of issuance: March 20, 2002.

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

Amendment Nos.: 158 and 120.

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

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44171). The November 15, 2001, letter 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 March 20, 2002.

No significant hazards consideration comments received: No.

National Aeronautics and Space Administration, Docket Nos. 50–30 and 50–185, the Plum Brook Test Reactor and the Plum Brook Mockup Reactor, Sandusky, Ohio

Date of application for amendments: December 20, 1999, as supplemented by letters dated March 26, November 19, and December 20, 2001, and January 24, 2002.

Brief description of amendments: The amendment allows decommissioning of the PBRF in accordance with NASA's application as supplemented. Pursuant to 10 CFR 50.82(b)(5), the approved decommissioning plan will be a supplement to the Safety Analysis Report or equivalent.

Date of issuance: March 20, 2002.

Effective date: March 20, 2002.

Amendment Nos.: Amendment No. 11 to Plum Brook Test Reactor and Amendment No. 7 to the Plum Brook Mockup Reactor.

Facility Operating License Nos. TR–3 and R–93: These amendments consist of changes to the Facility Licenses.

*Date of initial notice in **Federal Register:*** January 22, 2002 (67 FR 2924). The January 24, 2002, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation enclosed with the amendments dated March 20, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 20, 2001, as supplemented January 28 and February 21, 2002.

Brief description of amendment: The amendment revised the Technical Specifications, Section 2.1.1.2, to reflect the results of cycle-specific calculations performed for the upcoming Operating Cycle 9, and Section 5.6.5.b, to delete two redundant references.

Date of issuance: March 13, 2002.

Effective date: As of the date of issuance, to be implemented prior to startup from Refueling Outage 8.

Amendment No.: 105.

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

*Date of initial notice in **Federal Register:*** December 26, 2001 (66 FR 66468). The licensee's January 28 and February 21, 2002, supplemental letters provided clarifying information that was within the scope of the amendment request and did not change the initial proposed no significant hazards consideration determination.

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 26, 2001.

Brief description of amendment: The amendment revises Table 3.6.1.3–1, ''Secondary Containment Bypass Leakage Paths Leakage Rate Limits,'' to reflect the NRC staff's approval of the licensee's proposed modification of two primary containment isolation valves on feedwater piping from air-operated to become simple check valves.

Date of issuance: March 8, 2002.

Effective date: As of the date of issuance to be implemented prior to startup from Refueling Outage 8.

Amendment No.: 104.

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

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5329).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

North Atlantic Energy Service Corporation, et al., Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: February 28, 2001, as supplemented July 31, 2001, and December 21, 2001.

Description of amendment request: The amendment changes Seabrook Station Technical Specification 3/4.8.1.1 A.C. Sources—Operating. The changes are related to allowed outage time for restoration or verification of the operability of offsite power sources and to emergency diesel generator surveillance requirements.

Date of issuance: March 7, 2002.

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

Amendment No.: 80.

Facility Operating License No. NPF–86: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 18, 2001 (66 FR 20007). The July 31, 2001, and December 21, 2001, letters were within the scope of and did not affect the staff's finding of no significant hazards considerations.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 7, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 30, 2001, as supplemented September 7, October 16, and December 5, 2001, and January 18, 2002.

Brief description of amendments: The amendments revised Technical Specification 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to allow a one-time deferral of the Type A containment integrated leakage rate test (ILRT) at the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The Unit 1 test may be deferred to no later than May 3, 2007, and the Unit 2 test may be deferred to no later than October 30, 2007, resulting in an extended interval of 15 years for performance of the next ILRT at each unit. Additionally, the amendments allow a one-time deferral of the drywell-to-suppression chamber bypass leakage test, Surveillance Requirement (SR) 3.6.1.1.2, so that it will continue to be conducted along with the ILRT.

Date of issuance: March 8, 2002.

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

Amendment Nos.: 202, 176.

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

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5330). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: August 7, 2001.

Brief description of amendment: This amendment adds a response time requirement to the Technical Specifications for the Source Range Neutron Flux Reactor Trip function.

Date of issuance: March 8, 2002.

Effective date: March 8, 2002.

Amendment No.: 157.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5332). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: June 19, 2001.

Brief description of amendment: This amendment approves inclusion of two upgraded 7300 Process Protection System instrument cards (NLP—Loop Power Supply and Isolator card, and NSA—Summing Amplifier card) into the response time testing elimination population. The associated Bases for Technical Specification 3/4.3.1 is being revised to reflect this change.

Date of issuance: March 12, 2002.

Effective date: March 12, 2002.

Amendment No.: 158.

Facility Operating License No. NPF–12: Amendment revises the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38766).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 12, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 9, 2002.

Brief description of amendments: The amendments revise the Technical Specification 5.4, ''Technical Specifications (TS) Bases Control'' to delete the term ''unreviewed safety question.''

Date of issuance: March 19, 2002.

Effective date: March 19, 2002, to be implemented within 60 days of issuance.

Amendment Nos.: Unit 2–184; Unit 3–175.

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

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5333). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 2002.

No significant hazards consideration comments received: No.

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: December 14, 2001.

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

Date of issuance: March 8, 2002.

Effective date: As of the date of issuance and shall be implemented by August 1, 2002.

Amendment Nos.: 228/170.

Facility Operating License Nos. DPR–57 and NPF–5: Amendments revise the Technical Specifications and associated Bases.

Date of initial notice in Federal Register: February 5, 2002 (67 FR 5333). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: April 27, 2001.

Brief description of amendments: The amendments revised the Technical Specifications 3.3.6, ''Containment Ventilation Isolation Instrumentation,'' to extend the surveillance test interval for Potter and Brumfield type motor-driven slave relays in the containment ventilation isolation system from 92 days to 18 months. The associated Bases for SR 3.3.6.5 will be revised to reflect this change.

Date of issuance: February 21, 2002.

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

Amendment Nos.: 124/102.

Facility Operating License Nos. NPF–68 and NPF–81: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 21, 2001 (66 FR 31714). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 21, 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: May 30, 2001.

Brief description of amendments: The proposed amendment permits relaxation of the allowed outage times and bypass test times for limiting conditions for operation outlined in Technical Specifications 3.3.1, ''Reactor Trip System Instrumentation,'' and 3.3.2, ''Engineered Safety Features Actuation System Instrumentation.''

Date of issuance: March 19, 2002.

Effective date: The amendments are effective as of the date of issuance, and shall be implemented within 30 days of the day of issuance.

Amendment Nos.: Unit 1–136; Unit 2–125.

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

*Date of initial notice in **Federal Register:*** August 22, 2001 (66 FR 44177). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 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: August 2, 2001.

Brief description of amendments: The amendments consist of revision to Technical Specifications 3/4.6.1.6 regarding containment structural integrity.

Date of issuance: March 19, 2002.

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

Amendment Nos.: Unit 1–137; Unit 2–126.

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 2929). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 2002.

No significant hazards consideration comments received: No.

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: March 12, 2001.

Brief description of amendments: The amendments delete Sequoyah Technical Specification (TS) Surveillance Requirement 4.7.7.a from TS 3/4.7.7, ''Control Room Emergency Ventilation Systems,'' and adds a new Section 3/4.7.13, ''Control Room Air-Conditioning System (CRACS),'' to the TS. This TS addition will also provide the necessary requirements, consistent with NUREG–1431, to address the condition when main control room chillers and air handling units are inoperable.

Date of issuance: February 27, 2002.

Effective date: February 27, 2002.

Amendment Nos.: 273 and 262.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** April 18, 2001 (66 FR 20011). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 27, 2002.

No significant hazards consideration comments received: No.

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: January 15, 2002 (TS 01–13).

Brief description of amendments: The amendments revised Technical Specifications (TSs) Section 4.0.5.c to provide an exception to the recommendations of Regulatory Position c.4.b NRC Regulatory Guide 1.14, Revision 1, ''Reactor Coolant Pump Flywheel Integrity,'' dated August 1975. The exception allows either (a) a qualified in-place ultrasonic volumetric examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or (b) a surface examination (magnetic particle testing and/or liquid penetrant testing) of exposed surfaces of the removed flywheel to be conducted at approximately 10-year intervals.

Date of issuance: March 8, 2002.

Effective date: Date of issuance, to be implemented within 45 days of issuance.

Amendment Nos.: 274/263.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revised the TSs.

Date of initial notice in ***Federal Register:*** February 5, 2002 (67 FR 5339). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50–339, North Anna Power Station, Unit 2, Louisa County, Virginia

Date of application for amendment: January 9, 2001.

Brief description of amendment: This amendment revises the Facility Operating License (FOL) to remove expired license conditions, make editorial changes in the FOL, relocate license conditions, and remove license conditions associated with completed modifications.

Date of issuance: March 19, 2002.

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

Amendment Nos.: 211.

Facility Operating License No. NPF–7: Amendment changes the FOL.

Date of initial notice in ***Federal Register:*** February 21, 2001 (66 FR 11065). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 19, 2002.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: April 11, 2000, as supplemented August 28, and November 20, 2000, April 11, July 31, November 19, and December 20, 2001, and February 8, 2002.

Brief Description of amendments: These amendments revise the Technical Specifications requirements to be consistent with an alternative source term in accordance with the requirements of 10 CFR 50.67, ''Accident Source Term.''

Date of issuance: March 8, 2002.

Effective date: March 8, 2002.

Amendment Nos.: 230 and 230.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the Technical Specifications.

Date of initial notice in ***Federal Register:*** June 27, 2001 (66 FR 34289). The supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 8, 2002.

No significant hazards consideration comments received: No.

Yankee Atomic Electric Co., Docket No. 50–29, Yankee Nuclear Power Station (YNPS) Franklin County, Massachusetts

Date of application for amendment: September 28, 2000, as supplemented by letters dated October 12, 2000, April 18, May 29 and June 28, 2001, and March 4, 2002.

Brief description of amendment: The amendment revises License Condition 2.C.(3) to reference the revisions of the Physical Security Plan, Guard Training and Qualification Plan, and Safeguards Contingency Plan which provide for movement of the spent nuclear fuel from the spent fuel pool to the Independent Spent Fuel Storage Installation.

Date of issuance: March 13, 2002.

Effective date: March 13, 2002.

Amendment No.: 156.

Facility Operating License No. DPR–3: The amendment revised the License.

Date of initial notice in ***Federal Register:*** March 26, 2001 (66 FR 16501). The April 18, May 29, and June 28, 2001, and March 4, 2002, supplemental letters provided additional clarifying information that did not expand the scope of the application as originally noticed and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 2002.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 25th day of March, 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–7799 Filed 4–1–02; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549.

Extension

Rule 17a–11 SEC File No. 270–94; OMB Control No. 3235–0085

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') is soliciting comments on the collection of information summarized below. The Commission plans to submit this existing collection of information to the Office of Management and Budget for extension and approval.

Rule 17a–11 (17 CFR 240.17a–11) requires broker-dealers to give notice when certain specified events occur. Specifically, the rule requires a broker-dealer to give notice of a net capital deficiency on the same day that the net capital deficiency is discovered or a broker-dealer is informed by its designated examining authority or the Commission that it is, or has been, in violation of its minimum requirement under Rule 15c3–1 (17 CFR 240.15c3–1) of the Securities Exchange Act of 1934 (''Exchange Act''). Under Rule 17a–11 an over-the-counter (''OTC'') derivatives dealers must also provide notice to the Commission when a net capital deficiency is discovered but need not give notice to any SRO because OTC derivatives dealers are only required to register with the Commission.

Rule 17a–11 also requires a broker-dealer to send notice promptly (within 24 hours) after the broker-dealer's aggregate indebtedness is in excess of 1,200 percent of its net capital, its net capital is less than 5 percent of aggregate debit items, or its total net capital is less than 120 percent of its required minimum net capital. In addition, a broker-dealer must give notice if it fails to make and keep current books and records required by Rule 17a–3 (17 CFR 240.17a–3), if any material inadequacy is discovered as defined in Rule 17a–5(g) (17 CFR 240.17a–5(g)), and if back testing exceptions are identified pursuant to Appendix F of Rule 15c3–1 (17 CFR 240.15c3–1f) for a broker-dealer registered as an OTC derivatives dealer.

The notice required by the rule alerts the Commission, self-regulatory organizations (''SROs''), and the Commodity Futures Trading Commission (''CFTC'') if the broker-dealer is registered as a futures commission merchant, which have oversight responsibility over broker-dealers, to those firms having financial or operational problems.

Because broker-dealers are required to file pursuant to Rule 17a–11 only when certain specified events occur, it is difficult to develop a meaningful figure for the cost of compliance with Rule 17a–11. In 2001, the Commission received 692 notices under this rule from 627 broker-dealers. Each broker-dealer reporting pursuant to Rule 17a–11 will spend approximately one hour preparing and transmitting the notice as required by the rule. Accordingly, the total estimated annualized burden for 2001 was 692 hours. With respect to those broker-dealers that must give notice under Rule 17a–11, the Commission staff estimates that the approximate administrative cost, consisting mostly of accountant clerical work, to broker-dealers would be $24.53 per hour (based on the Securities Industry Association salary survey and including 35% in overhead costs). Therefore, based on approximately one hour per notice and a total of 692 notices filed, the total annual expense for the reporting broker-dealers in 2001 was approximately $16,975.

Written comments are invited on: (a) Whether the proposed collection of information is necessary for the proper performance of the functions of the agency, including whether the information will have practical utility; (b) the accuracy of the agency's estimate of the burden of the collection of information; (c) ways to enhance the quality, utility, and clarity of the information collected; and (d) ways to minimize the burden of the collection of information on respondents, including through the use of automated collection techniques or other forms of information technology. Consideration will be given to comments and suggestions submitted in writing within 60 days of this publication.

Please direct your written comments to Michael E. Bartell, Associate Executive Director, Office of Information Technology, Securities and Exchange Commission, 450 5th Street, NW, Washington, DC 20549.

Dated: March 26, 2002.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 02–7866 Filed 4–1–02; 8:45 am]

BILLING CODE 8010–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration on the Chicago Stock Exchange, Inc. (BellSouth Corporation, Common Stock, $1.00 Par Value) File No. 1–8607

March 27, 2002.

BellSouth, a Georgia corporation (''Issuer''), has filed an application with the Securities and Exchange Commission (''Commission''), pursuant to section 12(d) of the Securities Exchange Act of 1934 (''Act'')[1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Common Stock, $1.00 par value (''Security''), from listing and registration on the Chicago Stock Exchange, Inc. (''CHX'' or ''Exchange'').

The Issuer stated in its application that it has complied with the rules of the CHX that govern the removal of securities from listing and registration on the Exchange. In making the decision to withdraw the Security from listing and registration on the CHX, the Issuer considered the direct and indirect cost associated with maintaining multiple listing. The Issuer stated in its application that the Security has been listed on the New York Stock Exchange, Inc. (''NYSE'') since the company began operations in 1983. The Issuer represented that it will maintain its listing on the NYSE.

The Issuer's application relates solely to the Security's withdrawal from listing on the CHX and shall not affect its listing on the NYSE or its registration under section 12(b) of the Act.[3]

Any interested person may, on or before April 19, 2002 submit by letter to the Secretary of the Securities and Exchange Commission, 450 Fifth Street, NW, Washington, DC 20549–0609, facts bearing upon whether the application has been made in accordance with the

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).

[3] 15 U.S.C. 78*l*(b).