the NRC home page site for 60 days after the signature date of this notice.

Comments and questions should be directed to the OMB reviewer listed below by May 30, 2002. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date. Bryon Allen, Office of Information and Regulatory Affairs (3150–0158), NEOB–10202, Office of Management and Budget, Washington, DC 20503.

Comments can also be submitted by telephone at (202) 395–3087.

The NRC Clearance Officer is Brenda Jo. Shelton, 301–415–7233.

Dated at Rockville, Maryland, this 23rd day of April 2002.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 02–10591 Filed 4–29–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–368]

Entergy Operations, Inc.; Notice of Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 244 to Facility Operating License No. NPF–6 issued to Entergy Operations, Inc. (the licensee), which revised the Operating License and Technical Specifications (TSs) for operation of Arkansas Nuclear One, Unit 2, located in Pope County, Arkansas. The amendment is effective as of the date of issuance.

The amendment modified the Operating License and the TSs to allow an increase in the maximum authorized reactor core power level from 2815 megawatts thermal (MWt) to 3026 MWt, which represents a power increase of about 7.5 percent and is considered to be an extended power uprate. Also, operation at the uprated power requested by the proposed amendment resulted in increases in dose consequences for certain postulated accidents considered in the accident analyses in the Safety Analysis Report; however, the doses remained within the regulatory limits. In addition, although unrelated to the proposed power uprate, the proposed amendment clarified portions of the control element assembly TSs.

The application for the amendment 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 and Opportunity for a Hearing in connection with this action was published in the **Federal Register** on December 27, 2001 (66 FR 66945). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (67 FR 20176, published April 24, 2002).

Further details with respect to the action see (1) the application for amendment dated December 19, 2000, as supplemented by letters dated May 30, June 20, 26 (two letters), 27, and 28, July 3 and 24 (two letters), August 7, 13, 21, 23, and 30, September 14, October 1, 12 (two letters), 17, 30 (two letters), and 31, November 9, 16 (three letters), and 17, and December 5, 6 (two letters), 10, and 20, 2001, and January 14, 15, and 31, February 7 (two letters), and March 1, 2002, (2) Amendment No. 244 to License No. NFP–6, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically 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/NRC/ADAMS/index.html.* Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contract the NRC Public Document Room Reference staff by telephone at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Dated at Rockville, Maryland, this 24th day of April 2002.

For the Nuclear Regulatory Commission.

Thomas W. Alexion,

Project Directorate IV, Project Manager, Section 1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 02–10589 Filed 4–29–02; 8:45 am]

BILLING CODE 7590–01–P

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 April 5, 2002 through April 18, 2002. The last biweekly notice was published on April 16, 2002 (67 FR 18641).

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 30, 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/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 304–415–4737 or by e-mail to *pdr@nrc.gov.*

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2), Darlington County, South Carolina

Date of amendment request: February 21, 2002.

Description of amendment request: The proposed amendment to the Technical Specifications for HBRSEP2 will modify the containment vessel spray nozzle testing frequency specified in the Surveillance Requirement 3.6.6.8 from ''10 years'' to ''following activities which could result in nozzle blockage.''

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:

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

The proposed change revises the Surveillance frequency from once per [''}10 years[''] to [''}following activities [which] could result in nozzle blockage.[''] The Containment Spray System is not considered as an initiator of any analyzed accident. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed accident. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated is not significantly increased.

The proposed change revises the Surveillance frequency. Reduced testing is acceptable where operating experience has shown that components routinely pass the Surveillance when performed at the specified interval. The system design and construction materials provide assurance that the production of significant corrosion products is unlikely. Since activities that could introduce foreign material are the most likely cause for obstruction, testing or inspection following such an activity would verify that the nozzles are unobstructed and capable of performing their safety function. Such events would necessarily involve a substantive breakdown in foreign material controls during such activities. As a result, the consequences of any accident previously evaluated are not significantly affected.

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 to the test frequency for the Containment Spray System nozzles does not involve the use or installation of new equipment. Currently 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.

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 margin between containment pressure response and containment design pressure will not be affected because the design and functioning of the Containment Spray System is unchanged. Since the system is not susceptible to corrosion induced obstruction, nor is the introduction of foreign material from the exterior likely, the proposed surveillance frequency is sufficient to provide high confidence that the Containment Spray System will be available to provide the flow necessary in the event that the safety function is required. Therefore, the capacity of the system will remain unchanged.

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

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

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

NRC Section Chief: Thomas Koshy, Acting.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: March 13, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, to permit selective implementation of an alternative source term (AST). The proposed amendment would modify the TS requirements for movement of irradiated fuel and performing core alterations.

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

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

Implementation of the Alternative Source Term does not affect the design or operation of HBRSEP, Unit No. 2; rather, once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences of the postulated accident. A review of the HBRSEP, Unit No. 2, Updated Final Safety Analysis Report (UFSAR) shows that the components and systems affected by the proposed changes are not initiators of any previously analyzed accident. Therefore, there is no significant increase in the probability of any previously analyzed accident.

The implementation of the Alternative Source Term has been evaluated in a revision to the HBRSEP, Unit No. 2, Fuel Handling Accident. Based on the results of this analysis, it has been demonstrated that, with the requested changes to the Technical Specifications, the dose consequences of a postulated Fuel Handling Accident are within the regulatory guidance provided by the NRC for use with the Alternative Source Term. This guidance is presented in 10 CFR 50.67 and Regulatory Guide 1.183. Since automatic actuation of the control room emergency filtration system, automatic actuation of containment ventilation isolation, containment penetration operability, and the containment purge filter system are not credited in the revised analysis for the Fuel Handling Accident, eliminating these requirements during the movement of irradiated fuel assemblies will not result in a significant increase in the consequences of any previously evaluated accident. In addition, a review of the HBRSEP, Unit No. 2, UFSAR shows that the only accident resulting in dose consequences that is postulated to occur during core alterations is the Fuel Handling Accident. Therefore, the Applicability changes associated with core alterations will not result in a significant increase in the consequences of any previously evaluated accident.

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

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

The proposed changes are supported by the revised design basis Fuel Handling Accident analysis. The proposed changes do not introduce any new modes of plant operation and do not involve physical modifications to the plant.

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

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

The proposed changes are associated with the implementation of a new licensing basis for HBRSEP, Unit No. 2. The new licensing basis implements an Alternative Source Term in accordance with 10 CFR 50.67 and the associated Regulatory Guide 1.183. The results of the revised Fuel Handling Accident analysis, revised in support of this submittal, are subject to revised acceptance criteria. This analysis has been performed using conservative methodologies in accordance with the regulatory guidance. The dose consequences of the limiting Fuel Handling Accident are within the acceptance criteria also found in the regulatory guidance associated with Alternative Source Terms.

The proposed changes continue to ensure that doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Specifically, the margin of safety for this accident is considered to be that provided by meeting the applicable regulatory limits, which are conservatively set below the 10 CFR 50.67 limits. With respect to control room personnel doses, the margin of safety (the difference between the 10 CFR 50.67 limits and the regulatory limits defined by 10 CFR 50, Appendix A, Criterion 19 (GDC–19)) continues to be satisfied.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, they do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Thomas Koshy, Acting

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

Date of amendment request: March 26, 2002

Description of amendment request: The proposed amendments would modify the Technical Specifications definitions for Engineered Safety Feature Response Time and Reactor Trip System Response Time to provide for verification of response time for selected instruments provided that the instruments and methodology for verification have been previously reviewed and approved by the staff.

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

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

No. The proposed amendment to the Technical Specifications does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The same reactor trip system (RTS) and engineered safety features actuation system (ESFAS) instrumentation is used, and the response time allocation/modeling assumptions in UFSAR [updated final safety analysis report] Chapter 15 analysis remain unchanged. Only the methodology of time response verification is changed. The proposed change will not result in the modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. Therefore, the proposed amendment does not result in the increase in the probability or consequences of an accident previously evaluated.

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

No. This change does not alter the performance of the reactor protection system (RPS) or ESFAS systems. All RPS and ESFAS channels will still have response time verified by test before placing the channel in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain RPS and ESFAS channels (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the channel characteristic. Implementation of the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

No. This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method is modified to allow for the use of actual test or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect channel response time. Based on the above, it is concluded that the proposed license amendment request does not result in a reduction in margin with respect to plant safety.

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

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

NRC Section Chief: John A. Nakoski

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

Date of amendment request: March 20, 2002

Description of amendment request: The proposed amendment revises the reporting requirements specified in Section 2.E of the Facility Operating License and Technical Specification Section 5.6.4 by eliminating requirements that provide the U.S. Nuclear Regulatory Commission (NRC) with information that is not risk significant, and changes the reporting time period to be consistent with Section 50.73 of Title 10 of the Code of Federal Regulations (10 CFR).

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.

These changes involve administrative requirements only. The plant's design basis and the Updated Safety Analysis Report accident analysis are not affected. In addition, none of these reporting requirements support the plant's emergency plan. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change deletes non-risk significant reporting requirements and does not affect plant design or operation.

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.

This change only impacts administrative reporting requirements. It does not impact the design or operation of any plant system, structure, or component. In addition, no Technical Specification Safety Limit or instrument allowable value are affected. 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: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005

NRC Section Chief: Robert A. Gramm

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

Date of amendment request: March 13, 2002

Description of amendment request: The proposed amendment would 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 would be 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 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 March 13, 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.

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., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm

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

Date of amendment request: March 13, 2002.

Description of amendment request: The proposed amendment would correct several errors that were found subsequent to Nuclear Regulatory Commission (NRC) issuance of Amendment 215, which converted the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO–1), to Improved TSs.

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

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

The non-administrative proposed changes describe required actions to be taken upon loss of required electrical equipment, the number of non-licensed operators required on-site in Modes 1, 2, 3, and 4, the limitations on radiological effluent releases, and a clarification of automatic isolation capabilities when the control room is operating in the emergency recirculation mode. The proposed changes are not considered accident initiators nor do they result in a change to the physical characteristics of plant equipment. The proposed changes act to provide reasonable response to lost equipment, defense in depth, limitations on radiological release, and placing equipment in a fail-safe condition and do not adversely affect the accident analysis.

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

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

The non-administrative proposed changes describe required actions to be taken upon loss of required electrical equipment, the number of non-licensed operators required on-site in Modes 1, 2, 3, and 4, the limitations on radiological effluent releases, and a clarification of automatic isolation capabilities when the control room is operating in the emergency recirculation mode. None of the proposed changes can initiate any type of accident in and by themselves. The proposed changes act only to provide reasonable response to lost equipment, defense in depth, limitations on radiological release, and placing equipment in a fail-safe condition.

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

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

The aforementioned non-administrative proposed changes do not significantly impact the margin of safety. The proposed required action to be performed upon loss of a required inverter in Modes 5 and 6 provides a possible alternative to suspending fuel movement by taking conservative action to identify and declare inoperable, all equipment affected by the loss of the required inverter. These actions may in turn, require suspension of refueling activities, but in any event act to ensure the unit is maintained in a safe shutdown condition.

The increase in the number of required non-licensed operators on-site in Modes 1, 2, 3, and 4 is conservative and acts to improve the margin to safety by expanding the station's defense in depth.

Complying with the radiological effluent release limitations as set forth in 10 CFR [part] 20 (prior to revision) provides acceptable assurance that the health and safety of the public will be maintained. ANO–1 current complies with this version of 10 CFR [part] 20.

Finally, the proposed Bases change for the CREVS clarifies that automatic isolation signals and devices are not required when the control room is already isolated and operating in the emergency recirc[ulation] mode of operation. When in this configuration, the control room habitability system meets the safety function for which it was designed. Thus, this clarification does not affect the margin to safety.

Therefore, the proposed changes 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 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: February 19, 2002.

Description of amendment request: Entergy Operations, Inc. (EOI) requests a license amendment for the Grand Gulf Nuclear Station, Unit 1. EOI proposes to amend Technical Specification (TS) 3.8.1, ''AC [Alternating Current] Sources—Operating'' to remove the reactor operational MODE restrictions for testing the High Pressure Core Spray (HPCS) Diesel Generator 13 (DG 13). The proposed change would remove the restriction associated with surveillance requirements (SRs) that prohibit performing the required DG 13 testing while the plant is in reactor operation MODES 1, 2, or 3. This TS change would allow the performance of all SRs for DG 13 during any MODE of plant 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 HPCS DG and its associated emergency loads are accident mitigating features, not accident initiating equipment. Therefore, there will be no impact on any accident probabilities by the approval of the requested amendment.

The design of plant equipment is not being modified by these proposed changes. As such, the ability of the DG to respond to a design basis accident will not be adversely impacted by these proposed changes. The capability of the DG to supply power in a timely manner will not be compromised by permitting performance of DG testing during periods of power operation. Additionally, limiting testing to only one DG at a time ensures that design basis requirement for backup power is met, should a fault occur on the tested DG. Therefore, there would be no significant impact on any accident consequences.

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 causal mechanisms would be created as a result of NRC [U.S. Nuclear Regulatory Commission] approval of this amendment request since no changes are being made to the plant that would introduce any new accident causal mechanisms. Equipment will be operated in the same configuration with the exception of the plant MODE in which the testing is currently conducted. This amendment request does not impact any plant systems that are accident initiators; neither does it adversely impact any accident mitigating systems.

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.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes to the testing requirements for the HPCS DG do not affect the operability requirements for the DG, as verification of such operability will continue to be performed as required. Continued verification of operability supports the capability of the DG to perform its required function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Consequently, the performance of these fission product barriers will not be impacted by implementation of this proposed amendment.

In addition, the proposed changes involve no changes to setpoints or limits established or assumed by the accident analysis. On this and the above basis, no safety margins will be impacted.

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: February 25, 2002.

Description of amendment request: Entergy Operations, Inc. (EOI) requests modification of the Grand Gulf Nuclear Station, Unit 1 Technical Specifications (TS) to add a new Special Operations Limiting Condition for Operation (LCO) (Suppression Pool Makeup—MODE 3). The new TS provision would allow installation of the Upper Containment Pool (UCP) reactor cavity gates, and draining the reactor cavity pool portion of the UCP while still in reactor operation MODE 3, ''Hot Shutdown,'' with the reactor pressure less than 230 pounds per square inch gauge. EOI also requests modification to the applicability of the UCP gates surveillance requirement, TS 3.6.2.4, ''Suppression Pool Makeup (SPMU) System,'' to allow installation of the UCP gates in reactor operation MODE 1, ''Power Operation,'' MODE 2, ''Startup,'' or MODE 3, ''Hot Shutdown.'' The proposed changes would allow earlier installation of the UPC gates, and allow draining of the reactor cavity pool portion of the UCP while holding the plant in MODE 3 to facilitate an earlier start of certain refueling outage work evolutions.

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 changes revise the required water levels in the UCP and suppression pool. 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. 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 UCP 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 level will be raised above the current high water limit for the proposed Special Operations LCO 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 reduced post-LOCA [loss of coolant accident] Containment pressure ensures that post-accident dose consequences with no fission product scrubbing by Containment Spray (CS) is bounded by the DBA [design basis accident] LOCA.

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

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

Response: No.

The proposed changes to the water level requirements for the UCP 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.

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

Response: No.

The proposed changes to the UCP 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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm

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

Date of amendment request: March 14, 2002.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) by extending the allowed outage time (AOT), or completion time, associated with an inoperable Emergency Core Cooling System accumulator. The proposed changes are based on the methodology described in Topical Report WCAP–15049-A, ''Risk-Informed Evaluation of an Extension to Accumulator Completion Times,'' Revision 1. In addition to the AOT extension, other changes would be incorporated to make the ''Emergency Core Cooling Systems'' TSs consistent with NUREG–1431, ''Standard Technical Specifications—Westinghouse Plants.'' Format and editorial changes are included as necessary to facilitate the revision of the TS text to conform to the current TS page format.

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

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

No. The proposed changes consist of extending allowed outage times for required accumulator Technical Specification actions, elimination of alarm surveillance requirements associated with the accumulators, verifying boron concentration and editorial changes. These changes are independent of the probability or consequences of accidents previously evaluated in either of the Beaver Valley Power Station (BVPS) Updated Final Safety Analysis Reports (UFSARs). Since the accumulators are not accident initiators, they do not affect the probability of accidents. An NRC [Nuclear Regulatory Commission] approved generic analysis for Westinghouse plants, which is applicable to BVPS, concludes that extending the accumulator allowed outage time for reasons other than boron concentration out of limit is acceptable because the impact of core damage frequency has been shown to be within acceptable limits. The extension to the allowed outage time for boron not being within limits is consistent with NUREG–1431 and acceptable because the boron is not assumed in the injection phase of a loss of coolant accident (LOCA).

The accumulators, however, do perform an accident mitigation function. Their mitigation function is also not affected by the proposed changes since none of the associated accident mitigation parameters are changed. The accumulator volume available for injection remains the same as before the proposed changes, as does the boron concentration of the contained water. The accumulator valve position requirement to be open with its power removed, and the nitrogen cover pressure limit are also not changed by this request. As a result the same amount of water, at the same boron concentration, will be injected into the Reactor Coolant System (RCS) in the same amount of time after the proposed changes are made as it was before the proposed changes. Due to the fact that the accident mitigation function of the accumulators is not affected by the proposed changes, the consequences of an accident previously evaluated is also not changed.

Since the duration of the allowed outage times is not an input into the safety analysis (i.e., the safety analysis assumes that all of the accumulators are operable), the extension of the allowed outage times has no impact on the safety analysis. Therefore, the proposed changes do 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?

No. Extending allowed outage times for required Technical Specification actions and eliminating alarm surveillance requirements associated with the accumulators would not affect the operation or maintenance of the accumulators. The accumulators will not be operating in any different manner following the proposed changes than they were before the proposed changes are made. They will not be subjected to any new environmental conditions or operational modes, or placed into any new configurations that could lead to any new failure mechanisms. The role of the accumulators following a LOCA is not altered by adopting the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated for BVPS.

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

No. The proposed changes do not involve any changes to accumulator parameters utilized in the accident analysis. There are no changes being made to the accumulator's water volume, boron concentration, nitrogen cover pressure or the position of the isolation valve. As a result, the assumptions made regarding the performance of the accumulators during an accident are unchanged. An NRC approved generic analysis for Westinghouse plants concludes that extending the accumulator allowed outage time for reasons other than boron concentration out of limit is acceptable because the impact on core damage frequency has been shown to be within acceptable limits. A plant specific risk assessment confirms that this generic analysis is applicable to BVPS. The extension to the allowed outage time for boron not being within limits is consistent with NUREG–1431 and acceptable because the boron is not assumed in the injection phase of a LOCA. Therefore, the proposed changes do not involve a significant reduction in a margin to safety.

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: March 27, 2002.

Description of amendment request: The licensee proposed to revise Section 3.6.3, ''Emergency Power Sources,'' of the Technical Specifications to extend the allowed outage time (AOT) for an inoperable diesel generator (DG) from the current 7 days to 14 days. In addition, Section 3.4.4, ''Emergency Ventilation System,'' and 3.4.5, ''Control Room Air Treatment System,'' would be revised to delete the Limiting Condition for Operation (LCO) that the DGs associated with operation of these systems be operable at all times.

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

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

No. The proposed amendment only affects the AOT for the DGs, and LCO for the emergency ventilation and control room air treatment systems. There will be no associated changes to the design, operational characteristics, function, or reliability of these systems.

These systems were designed to mitigate the consequences of various previously evaluated accidents and, as such, are not postulated to cause such accidents. Thus, the proposed amendment does not affect the safety function of these systems nor the credits given to these systems for mitigating accident consequences. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment does not affect accident initiators or precursors because it does not alter any design parameter, condition, equipment configuration, or manner in which the affected systems are operated. Further, it does not alter or prevent the ability of structures, systems, or components to perform their intended safety or accident mitigating functions. Accordingly, the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.

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

No. The proposed amendment does not change any design parameter, analysis methodology, safety limits or acceptance criteria. The revised requirements will continue to ensure reliability and operability of the affected systems. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Joel Munday, Acting.

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

Date of amendment request: March 28, 2002

Description of amendment request: The proposed amendment would change Technical Specification 3.0.3 to allow a longer period of time to perform a missed surveillance. The time would be 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 the specification: ''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 application dated March 28, 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.

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: L. Raghavan

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

Date of amendment request: March 29, 2002

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to change TS Section 5.5.12, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) to no later than September 2008. This would represent a one-time extension of the ILRT interval from 10 years to 15 years.

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

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

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident.

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

The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. The DAEC [Duane Arnold Energy Center] ILRT test history supports this conclusion. NUREG–1493, Performance-Based Containment Leak-Test Program, concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the DAEC's response to NRC Generic Letter 98–04 (''Potential for Degradation of the Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) after a Loss-of-Coolant Accident (LOCA) because of Construction and Protective Coating Deficiencies and Foreign Material in Containment'') serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing, thus maintaining containment leakage low. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage that may develop during power operation.

Therefore, the proposed Technical Specification change does not involve a significant increase in the 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 revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. Primary containment is designed to contain energy and fission products during and after an event. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. Revision to the Type A test interval does not change the events that could lead to containment failure. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident.

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

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

The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The proposed change involves only the extension of the interval between Type A containment leakage tests. The current interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1% to the individual risk and that increasing the Type A test interval would have minimal affect on this risk since about 95% of the potential leakage paths are detected by Type B and Type C testing. The DAEC and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the DAEC's response to NRC Generic Letter 98–04 serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing.

The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for the detection of gross containment leakage should it develop during power operation.

Therefore, the proposed Technical Specification change will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: March 18, 2002.

Description of amendment request: The proposed amendment would modify the Technical Specifications (TS) to remove the administrative requirement that a candidate for the plant operations manager position hold a Senior Reactor Operator License at the time of appointment. This proposed change to the TS endorses Regulatory Guide 1.8, Revision 3, ''Qualification and Training of Personnel for Nuclear Power Plants,'' and would, therefore, allow a broader base of qualified candidates to hold this position.

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 amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change provides enhancement to the current requirements and clarifies the qualification requirements for the operations manager position. This provides additional assurance that the personnel filling this position will be properly qualified. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change provides enhancement to the current requirements and clarifies the qualification requirements for the operations manager position. There are no structures, systems, or components affected by this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No

The proposed change does not involve a significant reduction in a margin of safety. The proposed change provides enhancement to the current requirements and clarifies the qualification requirements for the operations manager position. This provides additional assurance that the personnel filling this position will be properly qualified. 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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP 601 13th Street, NW., Suite 1000 South, Washington, DC 20005

NRC Section Chief: Richard J. Laufer.

Rochester Gas and Electric Corporation, Docket No. 50–244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of amendment request: March 18, 2002.

Description of amendment request: The proposed amendment would revise Surveillance Requirement (SR) 3.0.3 to extend the delay period, before entering a Limiting Condition for Operation LCO, following a missed surveillance. The delay period would be extended from the current limit of ''* * * up to 24 hours 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 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 March 18, 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 [a] Margin of Safety.

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

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

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

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: Daniel F. Stenger, Ballard Spahr Andrews & Ingersoll, LLP, 601 13th Street, NW., Suite 1000 South, Washington, DC 20005.

NRC Section Chief: Richard J. Laufer.

Southern Nuclear Operating Company, Inc, Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendment request: January 24, 2002.

Description of amendment request: The proposed amendments would delete requirements from the Technical Specifications (TS) (and, as applicable, other elements of the licensing bases) to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as described in NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the lessons learned from the accident that occurred at TMI, Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated July 2, 2001.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 14, 2002

Description of amendment request: The proposed amendment revises Technical Specifications (TS) 3.3.2 requirements for Loss of Power Instrumentation (Functional Unit 8) and the Technical Specifications 3.8.1.1, 3.8.1.2, and 3.8.1.3, for AC Sources.

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 changes do not change the plant design basis, system configuration or operation, and do not add or affect any accident initiator. Therefore, STPNOC concludes that there is no significant increase in the possibility of an accident previously evaluated.

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

Response: No

The proposed changes do not change the plant design basis, system configuration or operation, and do not add or affect any accident initiator. Therefore, STPNOC concludes the proposed change does not create the possibility of a new or different kind of accident previously evaluated.

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

Response: No

No actual plant equipment or accident analyses will be affected by the proposed change. Additionally, the proposed changes will not relax any criteria used to establish safety limits, will not relax any safety system settings, or will not relax the bases for any limiting conditions of operation. Therefore, STPNOC concludes the proposed changes 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 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 NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: December 7, 2001.

Description of amendment request: This proposed amendment revises Technical Specifications Surveillance Requirement 4.6.1.2, ''Containment Leakage.'' The proposed change will permit a one-time 5-year extension to the 10-year performance-based Type A test interval.

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

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

The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since extension of the containment Type A testing is not a physical plant modification that could alter the probability of accident occurrence nor, is an activity or modification by itself that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not result in a significant increase in the consequences of an accident as documented in NUREG–1493. The NUREG notes that very few potential containment leakage paths are not identified by Types B and C tests. It concludes that reducing the Type A (ILRT [integrated leak rate test]) testing frequency to once per twenty years leads to an imperceptible increase in risk.

North Anna provides a high degree of assurance through testing and inspection that the containment will not degrade in a manner detectable only by Type A testing. The last two Type A tests identified containment leakage within acceptable criteria, indicating a very leak-tight containment. Inspections required by the ASME [American Society of Mechanical Engineers] Code are also performed in order to identify indications of containment degradation that could affect leak-tightness. Separately, Types B and C testing, required by Technical Specifications, identifies any containment opening from design penetrations, such as valves, that would otherwise be detected by a Type A test. These factors establish that an extension to the North Anna Type A test interval will not represent a significant increase in the consequences of an accident.

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

The proposed revision to North Anna Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing does not create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure.

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

The proposed revision to North Anna Technical Specifications adds a one-time extension to the current interval for Type A testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to fifteen years from the last Type A test. The proposed extension to Type A testing will not significantly reduce the margin of safety. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributes about 0.1 percent of the overall risk and that decreasing the Type A testing frequency would have a minimal affect on this risk since 95% of the Type A detectable leakage paths would already be detected by Type B and C testing. Furthermore, for North Anna, maintaining the containment subatmospheric during plant operations further reduces the risk of any containment leakage path going undetected.

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

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: November 29, 2001.

Description of amendment request: This amendment proposes to revise the Technical Specifications containment air partial pressure versus service water temperature operating limits and surveillance requirements for the recirculation spray pump start delay times.

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 to the containment air partial pressure versus service water temperature operating curve and recirculation spray timer delays will continue to ensure that the containment remains operable to mitigate Design Basis Accidents. The revised containment operating curve and timer delays do not affect the probability of occurrence of any accident previously analyzed. The revised containment licensing basis analyses use approved analytical methods and continue to demonstrate that the established accident analysis acceptance criteria are met. Therefore, there is no increase in the probability or consequences of any 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 to the containment air partial pressure versus service water temperature operating curve and recirculation spray timer delays will not create any new accident or event initiators. The containment will continue to be operated in a similar manner. No systems, structures, or components are being physically modified such that design function is being altered. The proposed change does not alter the nature of events postulated in the UFSAR [Updated Final Safety Analysis Report] nor does it introduce any unique precursor mechanisms. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type than previously evaluated.

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

The proposed changes to the containment air partial pressure versus service water temperature operating curve and recirculation spray time delays and supporting analyses maintain the existing safety margins. The revised containment analyses demonstrate that current acceptance criteria continue to be satisfied. Therefore, the proposed changes do not result in a significant reduction in margin of safety.

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

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: February 26, 2002.

Description of amendment request: This proposed amendment revises the surveillance frequency of the quench spray and recirculation spray system nozzles from a time period of every 10 years to whenever maintenance is conducted that could contribute to nozzle blockage.

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

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

The proposed change revises the surveillance frequency from every 10-years to ''following maintenance that could result in nozzle blockage.'' Analyzed events are initiated by the failure of plant structures, systems, or components. The containment spray system is not considered as an initiator of any analyzed event. The proposed change does not have a detrimental impact on the integrity of any plant structure, system or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated is not significantly increased.

The proposed change revises the surveillance frequency. Reduced testing is justified where operating experience has shown that routinely passing a surveillance test performed at a specified interval has no apparent connection to overall component reliability. In this case, routine surveillance testing at the specified frequency is not connected to any activity which may initiate reduced component reliability and therefore, has been of limited value in ensuring component reliability. Thus, the proposed frequency change is not significant for a reliability standpoint. The proposed containment spray nozzle surveillance frequency has been established based on achieving acceptable levels of equipment reliability.

This change does not affect the plant design. Due to the plant design, the spray ring headers are maintained dry. Formation of significant corrosion products is unlikely. Due to their location at the top of the containment, introduction of foreign material from exterior to the headers is unlikely. Since maintenance that could introduce foreign material is the most likely cause for obstruction, testing or inspection following such maintenance would verify the nozzle(s) remain unobstructed and the system's continued capability to perform its safety function. As a result, the consequences of any accident previously evaluated are not significantly affected by the proposed change.

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

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

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

The margin of safety for this system is based on the capacity of the spray headers. The system is not susceptible to corrosion induced obstruction or obstruction from external sources to the system. Performance of maintenance on the spray ring header would now require evaluation of the potential for nozzle blockage and the need for a test or inspection. Consequently, the spray header nozzles should remain unblocked and available in the event that the safety function is required. Therefore, the capacity of the system would remain unaffected. Hence, this change does not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

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

Date of amendment request: February 11, 2002.

Description of amendment request: This requested amendment would revise Facility Operating License Number NPF–7 to permit Virginia Electric and Power Company to irradiate a lead test assembly (LTA) at North Anna Power Station, Unit 2 to an end-of-life assembly average burnup of about 70 GWD/MTU, with the lead rod average burnup in this assembly approaching 73 GWD/MTU. The accompanying requested exemptions from 10 CFR 50.44, 10 CFR 50.46, and Appendix K of 10 CFR 50 will be processed separately.

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 probability of occurrence or the consequences of an accident previously evaluated is not significantly increased. The Framatome lead test assembly is very similar in design to the Westinghouse fuel that comprises the remainder of the core. The reload core design for the North Anna cycle where this assembly will operate to high burnup will meet all applicable design criteria. The performance of the Emergency Core Cooling system will not be affected by the operation of the lead test assembly, and operation of the LTA to high burnup will not result in a change to the North Anna reload design and safety analysis limits. Operation of one Framatome LTA to high burnup will not result in a measurable impact on normal operating plant releases, and will not increase the predicted radiological consequences of accidents postulated in Chapter 15 of the North Anna UFSAR [Updated Final Safety Analysis Report]. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. The possibility for a new or different type of accident from any accident previously evaluated is not created. The Framatome lead test assembly is very similar in design (both mechanical and composition of materials) to the resident Westinghouse fuel. All design and performance criteria will continue to be met and no new single failure mechanisms will be created. The irradiation of this fuel assembly to high burnup does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. The margin of safety is not significantly reduced. The operation of one Framatome lead test fuel assembly to high burnup does not change the performance requirements of any system or component such that any design criteria will be exceeded. The normal limits on core operation defined in the North Anna Technical Specifications will remain applicable for the irradiation of this assembly to high burnup. Evaluations will be performed to confirm that safety analyses based on the resident Westinghouse fuel remain applicable for the core in which the high burnup assembly is irradiated. Therefore, the margin of safety as defined in the Bases to the North Anna Technical Specifications is not significantly reduced.

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

Attorney for licensee: Ms. Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by 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: June 21, 2001.

Brief description of amendment: The amendment replaces individual main steamline leakage limits with an aggregate leakage limit, revising technical specification surveillance requirement 3.6.1.3.9, which provides leakage rate limits applicable to the main steamline isolation valves.

Date of issuance: March 26, 2002.

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

Amendment No.: 145.

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

*Date of initial notice in **Federal Register:*** October 3, 2001 (66 FR 50464). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 26, 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: September 17, 2001.

Brief description of amendment: The amendment revises the test frequency for the containment spray nozzles from ''once per 10 years'' to ''following activities that could result in nozzle blockage.''

Date of issuance: March 28, 2002.

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

Amendment No.: 146.

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

*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 28, 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: July 5, 2001, as supplemented December 28, 2001, and March 1, 2002.

Brief description of amendment: The amendment relaxes operability requirements for primary containment, secondary containment systems, and the standby gas treatment system during the movement of irradiated fuel and during core alterations.

Date of issuance: April 3, 2002

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

Amendment No.: 147

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64286). The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 3, 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: May 21, 2001

Brief description of amendment: The amendment revises the actions required if the refueling equipment interlocks become inoperable. The additional request in the application to revise the frequency of the refueling equipment interlock inputs channel functional test from 7 to 31 days is not included in the issued amendment and will be addressed by separate correspondence.

Date of issuance: April 4, 2002.

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

Amendment No.: 148.

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

*Date of initial notice in **Federal Register:*** December 26, 2001 (66 FR 66463). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 4, 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 18, 2001, as supplemented September 7 and 28, October 17, 23, 26, and 31, November 8 (2 letters), 20, 21, 29, and 30, and December 5, 6, 7, 13 (2 letters), 20, 21, and 26, 2001, and January 8, 15, 16, and 24, and March 15, 22, and 29, 2002.

Brief description of amendment: The amendment would allow an increase in the licensed power from 2894 megawatts thermal (MWt) to 3473 MWt.

Date of issuance: April 5, 2002.

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

Amendment No.: 149.

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

Date of initial notice in ***Federal Register:*** February 1, 2002 (67 FR 5001). The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 5, 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: November 30, 2001.

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

Date of issuance: April 9, 2002.

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

Amendment No.: 150.

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

Date of initial notice in ***Federal Register:*** February 19, 2002 (67 FR 7411). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 9, 2002.

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: July 27, 2001, as supplemented on January 16 and February 26, 2002.

Brief description of amendments: The amendments add additional references to Technical Specification 5.6.5.b to allow the use of ZIRLO™ clad fuel rods in the Calvert Cliffs reactor cores.

Date of issuance: April 8, 2002.

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

Amendment Nos.: 251, 228.

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

Date of initial notice in ***Federal Register:*** September 5, 2001 (66 FR 46476). The January 16 and February 26, 2002, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: July 31, 2001, as supplemented by letter dated February 5, 2002.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to allow an extension of the three-year inspection interval of the reactor coolant pump flywheel volumetric examination to 10 years. In addition, the inspection interval requirement would be moved to the administrative controls section of the TSs.

Date of issuance: April 11, 2002.

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

Amendment No.: 241.

Facility Operating License No. NPF–6: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44167). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2002.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: October 30, 2001, as supplemented by letters dated February 25 and March 13, 2002.

Brief description of amendment: The license amendment request proposes changes to Arkansas Nuclear One, Unit 2 Technical Specification (TS) 3/4.4.9, ''Pressure/Temperature Limits,'' and TS 3.4.12, ''Low Temperature Overpressure Protection (LTOP) System.'' The primary changes are to update the existing pressure/temperature limits from 21 to 32 effective full power years and to include additional restrictions in the LTOP TSs.

Date of issuance: April 15, 2002.

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

Amendment No.: 242.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64294). The supplemental letters provided clarifying information that did not change the staff's proposed no significant hazards consideration determination or expand the application beyond the scope of the **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 15, 2002.

No significant hazards consideration comments received: No.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: October 2, 2001.

Brief description of amendment: The amendment revises Technical Specification 3.3.2.1 Table 3.3–4, ''Engineered Safety Feature Actuation System Instrumentation Trip Values,'' Functional Unit 7.b, ''Loss of Power, 460 volt Emergency Bus Undervoltage,'' by changing the referenced bus from the 460 volt (V) to the 480 V bus, by removing the trip setpoint, and by slightly increasing the range of allowable values for the degraded voltage setting and its associated time delay.

Date of issuance: April 16, 2002.

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

Amendment No.: 243.

Facility Operating License No. NPF–6: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: October 31, 2001 (66 FR 55015). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 16, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 30, 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: April 8, 2002.

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

Amendment Nos.: 192 and 186.

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

Date of initial notice in ***Federal Register***: February 19, 2002 (67 FR 7417). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 8, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 30, 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: April 8, 2002.

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

Amendment Nos.: 153 and 139.

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

Date of initial notice in ***Federal Register***: February 19, 2002 (67 FR 7417). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 8, 2002.

No significant hazards consideration comments received: No.

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 application for amendments: February 22, 2002.

Brief description of amendments: The amendments would relocate technical specifications (TSs) 3/4.9.6, ''Refueling Operations—Manipulator Crane Operability,'' and TSs 3/4.9.7, ''Refueling Operations—Crane Travel—Spent Fuel Storage Pool Building,'' with associated Bases to the D. C. Cook updated final safety analysis report.

Date of issuance: April 18, 2002.

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

Amendment Nos.: 267 and 248.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: March 19, 2002, (67 FR 12603). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 18, 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: August 2, 2001, as supplemented November 2, December 4, and December 19, 2001, and January 7, 2002.

Description of amendment request: The amendment modifies the Technical Specifications to allow a one-time extension of the Appendix J Type A test (containment integrated leakage rate test) interval from 10 years to 15 years.

Date of issuance: April 11, 2002.

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

Amendment No.: 82.

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

Date of initial notice in ***Federal Register:*** December 12, 2001 (66 FR 64298). The December 4 and December 19, 2001, and the January 7, 2002, supplements were clarifying in nature, did not change the scope of the original **Federal Register** notice, and did not affect the staff's original proposed finding of no significant hazards considerations. The November 2, 2001, supplement was considered in the staff's proposed finding of no significant hazards considerations.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 2002.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: November 2, 2001.

Brief description of amendment: The amendment changes Technical Specification Table 3.3.1–1, ''Reactor Protective System Instrumentation,'' Item 1, ''Variable High Power Trip [VHPT],'' by increasing the maximum allowable value for the VHPT from less than or equal to 106.5 percent rated thermal power (RTP) to less than or equal to 111 percent RTP.

Date of issuance: April 10, 2002.

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

Amendment No.: 208.

Facility Operating License No. DPR–20: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59510). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 10, 2002.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: January 17, 2002, as supplemented on March 8 and 22, 2002.

Brief description of amendment: The amendment revises Salem, Unit No. 2, Technical Specifications Section 6.8.4.f, and provides for an alternate method for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.54(o), and 10 CFR Part 50, Appendix J, Option B. Specifically, the amendment allows a one-time interval increase for the Salem, Unit No. 2, Type A, Integrated Leakage Rate Test from a maximum of a 10-year interval to a maximum 15-year interval.

Date of issuance: April 11, 2002.

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

Amendment No.: 232.

Facility Operating License No. DPR–75: This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 7, 2002 (67 FR 10450). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 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: April 18, 2001.

Brief description of amendment: This amendment revises volumetric air flow units for Technical Specifications (TS) 4.7.6.c.1, c.3, e.1, e.3, f, and g to identify standard air flow units expressed as standard cubic feet per minute. Volumetric air flow units for TS 4.6.3.b.1, b.2, c.1, and d, and TS 4.9.11.b.1, b.3, d.1, e, and f are being revised to identify actual air flow units and are expressed as actual cubic feet per minute.

Date of issuance: April 11, 2002.

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

Amendment No.: 159.

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

*Date of initial notice in **Federal Register:*** May 30, 2001 (66 FR 29361). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 11, 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 No. 50–321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of application for amendment: January 4, 2002, as supplemented by letter dated March 15, 2002.

Brief description of amendment: The amendment revised the Safety Limit Minimum Critical Power Ratio for single loop operation in theTechnical Specifications to reflect the results of a cycle-specific calculation for Unit 1 Cycle 21.

Date of issuance: April 5, 2002

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

Amendment No.: 229.

Facility Operating License No. DPR–57: Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** February 5, 2002 (67 FR 5333). The supplement dated March 15, 2002, provided clarifying information that did not change the scope of the January 4, 2002, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 5, 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: 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: April 8, 2002.

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

Amendment Nos.: 125 and 103.

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

*Date of initial notice in **Federal Register:*** March 5, 2002 (67 FR 10015). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 8, 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 22, 2001, as supplemented by letters dated January 21; February 5, 14, and 27; and March 4, 2002. The supplementary letters provided clarifications of the application dated August 22, 2001, and did not alter the NRC staff's conclusions regarding no significant hazards consideration determination.

Brief description of amendments: The proposed amendments revise the Technical Specifications to reflect a 1.4 percent increase in the reactor core thermal power level from 3,800 megawatts thermal (MWt) to 3,853 MWt.

Date of issuance: April 12, 2002.

Effective date: The amendments are effective as of the date of issuance, to be implemented within 60 days from the date of issuance for Unit 1 and 60 days from date of installation of Δ94 steam generators for Unit 2.

Amendment Nos.: Unit 1–138; Unit 2–127.

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

*Date of initial notice in **Federal Register:*** December 26, 2001 (66 FR 66472). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 12, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 11, 2000, as supplemented by letters dated May 30, June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, and January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002.

Brief description of amendment: These amendments replace, in their entirety, the current technical specifications with a set of improved technical specifications based on NUREG–1431, Revision 1, ''Standard Technical Specifications, Westinghouse Plants,'' dated April 1995.

Date of issuance: April 5, 2002.

Effective date: As of the date of issuance and shall be implemented no later than September 2, 2002.

Amendment Nos.: 231 and 212.

Facility Operating License Nos. NPF–4 and NPF–7: Amendments change the Technical Specifications and the Facility Operating Licenses.

Date of initial notice in ***Federal Register:*** February 26, 2002 (67 FR 8827). The February 27, March 7, March 18, and March 22, 2002 supplements contained clarifying information only, and did not change or expand the scope of the February 26, 2002, **Federal Register** notice. The March 26, 2002 supplement withdrew a beyond scope issue and reduced the scope of the **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 5, 2002.

Dated at Rockville, Maryland, this 23rd day of April 2002.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 02–10456 Filed 4–29–02; 8:45 am]

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PRESIDENT'S COUNCIL ON INTEGRITY AND EFFICIENCY

EXECUTIVE COUNCIL ON INTEGRITY AND EFFICIENCY

Senior Executive Service Performance Review Board Membership

AGENCY: President's Council on Integrity and Efficiency (PCIE) and Executive Council on Integrity and Efficiency (ECIE).

ACTION: Notice.

SUMMARY: This notice sets forth the names and titles of the current membership of the PCIE/ECIE Performance Review Board.

EFFECTIVE DATE: May 1, 2002

FOR FURTHER INFORMATION CONTACT: Individual Offices of (the) Inspector General.

SUPPLEMENTARY INFORMATION:

I. Background

The Inspector General's Act of 1978, as amended, has created independent audit and investigative units-Offices of (the) Inspector General-at 57 Federal agencies. In 1981, the President's Council on Integrity and Efficiency (PCIE) was established by Executive Order. Executive Order 12805 of May 11, 1992, reaffirmed the PCIE and established the Executive Council on Integrity and Efficiency (ECIE). Both councils are interagency committees chaired by the Office of Management and Budget's Deputy Director for Management. Their mission is to continually identify, review, and discuss areas of weakness and vulnerability in Federal programs and operations to fraud, waste, and abuse, and to develop plans for coordinated, Government-wide activities that address these problems and promote economy and efficiency in Federal programs and operations. PCIE members include the 29 Inspectors General appointed by the President; ECIE members include the 28 Inspectors General appointed by their respective agency heads.

II. PCIE Performance Review Board

Under 5 U.S.C. 4314(c) (1)–(5) and in accordance with regulations prescribed by the Office of Personnel Management, each agency is required to establish one or more Senior Executive Service (SES) performance review boards. The purpose of these boards is to review and evaluate the initial appraisal of a senior executive's performance by the supervisor, along with any recommendations to the appointing authority relative to the performance of the senior executive.

Mark W. Everson,

Controller/Office of Federal Financial Management.

The current members of the PCIE/ECIE Performance Review Board are as follows:

Members	Title
AGENCY FOR INTERNATIONAL DEVELOPMENT	
James R. Ebbitt	Deputy Inspector General.
Adrienne Rish	Assistant Inspector General for Investigations.
Michael G. Carrol	Assistant Inspector General for Management.
Robert S. Perkins	Assistant Inspector General for Legal Counsel.
Bruce Crandlemire	Deputy Assistant Inspector General for Audit.
DEPARTMENT OF COMMERCE	
Edward L. Blansitt	Deputy Inspector General.
Judith J. Gordon	Assistant Inspector General Systems Evaluation.
Elizabeth T. Barlow	Counsel to the Inspector General.
Jill A. Gross	Assistant Inspector General for Inspections and Evaluations.
DEPARTMENT OF DEFENSE	
Carol Levy	Assistant Inspector General for Investigations.
David A. Brinkman	Director, Audit Follow-up & Technical Support Directorate.
Alan W. White	Director, Investigative Operations Directorate.
David Crane	Director for Intelligence Review.
Thomas J. Bonnar	Deputy Assistant Inspector General for Investigations.
Patricia A. Brannin	Deputy Assistant Inspector General for Audit Policy and Oversight.
C. Frank Broome	Director for Departmental Inquiries.
Joel L. Leson	Director for Administration and Information Management.
DEPARTMENT OF EDUCATION	
Tom Carter	Assistant Inspector General for Audit Services.
Don Reid	Assistant Inspector General for Investigation Services.
Helen Lew	Deputy Assistant Inspector General for Audit Services.
DEPARTMENT OF HEALTH AND HUMAN SERVICES	
Joe Green	Assistant Inspector General for Public Health Service Audits.
Dennis J. Duquette	Deputy Inspector General for Management & Policy.
Lewis Morris	Assistant Inspector General for Legal Affairs.
D. McCarty Thornton	Deputy Inspector General for Legal Affairs.
DEPARTMENT OF JUSTICE	
Robert L. Ashbaugh	Deputy Inspector General.
Mary W. Demory	Senior Executive for Strategic Planning and Special Projects.
DEPARTMENT OF LABOR	
Colleen B. Callahan	Deputy Inspector General for Management.
Stephen J. Cossu	Deputy Inspector General for Labor Racketeering& Fraud Investigations.
José Ralls	Administrative Officer.
Sylvia Horowitz	Counsel to the Inspector General.
DEPARTMENT OF TRANSPORTATION	
Todd J. Zinser	Deputy Inspector General.
Alexis M. Stefani	Assistant Inspector General for Audits.
Thomas J. Howard	Deputy Assistant Inspector General for Maritime and Departmental Programs.
DEPARTMENT OF THE TREASURY	
Dennis S. Schindel	Deputy Inspector General.
Marla A. Freedman	Assistant Inspector General for Audit.
Michael C. Tarr	Assistant Inspector General for Investigations.
William H. Pugh, III	Deputy Assistant Inspector General for Audit (Financial Management).
Elizabeth M. Redman	Deputy Assistant Inspector General for Investigations.
Richard K. Delmar	Counsel to the Inspector General.
DEPARTMENT OF THE TREASURY—TREASURY INSPECTOR GENERAL FOR TAX ADMINISTRATION	
Pamela J. Gardiner	Deputy Inspector General for Audit.
Daniel R. Devlin	Assistant Inspector General for Audit (HQ Ops And Ex Org).
Gordon C. Milbourn	Assistant Inspector General for Audit (Small Business and Corporate Progs).
Scott E. Wilson	Assistant Inspector General for Audit (Info Sys. Prog.).
Robert C. Cortesi	Deputy Inspector General for Investigations.