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 alist 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.

A request for a hearing and petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, 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 Mr. David E. Blabey, attorney for the licensee, 1633 Broadway, New York, New York 10019.

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

If a request for a hearing is received, the Commission's staff may issue the amendment after it completes its technical review and prior to the completion of any required hearing if it publishes a further notice for public comment of its proposed finding of no significant hazards consideration in accordance with 10 CFR 50.91 and 50.92. For further details with respect to the proposed action, see the licensee's application dated March 31, 1999, as supplemented by letters dated May 20, June 1, July 14, October 14, 1999, February 11, April 4, April 13, June 30, July 31, September 12, September 13, October 23, 2000, February 7, February 20, May 31, and August 6, 2001. 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. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to *pdr@nrc.gov.*

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 21st day of November 2001.

Guy S. Vissing,

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

[FR Doc. 01–29585 Filed 11–27–01; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

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

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.)

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 November 5 through November 16, 2001. The last biweekly notice was published on November 14, 2001 (66 FR 57116).

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 Public Document Room, 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 December 28, 2001, 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 Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. 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.* 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 Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, 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 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 Assess and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index.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, 304–415–4737 or by email to pdr@nrc.gov.

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

Date of amendment request: April 4, 2001, as supplemented on October 12, 2001.

Description of amendment request: The proposed amendment request would delete Technical Specifications (TSs) 5.3.1.B and 5.3.1.C. These TSs restrict the handling of heavy loads over irradiated fuel stored in the storage pool. The basis for deleting these TSs is the upgrade of the reactor building crane and associated handling systems to a single-failure proof system.

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

The proposed amendment does not:

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

Until August 2000, the reactor building crane was not single-failure-proof. For heavy load handling associated with the spent fuel pool, Oyster Creek was consistent with Section 5.1.4(2) of NUREG–0612: ''The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 are satisfied.'' An alternative to this is Section 5.1.4(1): ''The reactor building crane, and associated lifting devices used for handling of heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.'' The upgraded crane and handling systems satisfy the guidelines of Section 5.1.6. Therefore, the licensing basis for the reactor building crane with regard to its use in handling heavy loads above the spent fuel storage pool is being revised to include Section 5.1.4(1) of NUREG–0612 in addition to 5.1.4(2).

The cask drop protection system was required with the original crane because the load drop analysis will yield unacceptable consequences to the spent fuel storage pool (SFSP) structure. The cask drop protection system (CDPS) serves to mitigate the consequences of a cask drop accident involving the original crane which complied with NUREG–0612 Phase I. The upgraded single-failure-proof crane satisfies the criteria of NUREG–0612 Section 5.1.6. Therefore, the reactor building crane eliminates reliance on the design function of the CDPS because the probability of a heavy load drop is very low.

With the proposed revisions to the TSs, the evaluation criteria of NUREG–0612, Section 5.1 is met with a single-failure-proof crane that satisfies the guidelines of Section 5.1.6 or with consequence analyses that satisfies Section 5.1.4(2).

The proposed TS revisions do not significantly change the potential for unacceptable consequences to the plant in conducting heavy load handling above the SFSP because the probability of a load drop accident caused by use of the reactor building crane has been reduced to where it is very unlikely, and therefore, can be considered not credible within regulatory accepted standards.

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

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

The CDPS was installed in the Oyster Creek SFSP to mitigate the effects of a cask drop when the reactor building crane was not single-failure proof. The CDPS acts as a hydraulic dashpot to limit the velocity of a falling cask to attenuate impact forces to within acceptable levels. The CDPS structure cannot be removed from the spent fuel pool without eliminating its functional requirement. The use of the CDPS increases the duration of cask lifts and exposure to personnel. Therefore, eliminating the complications caused by the use of the CDPS together while improving the reliability of the crane and associated systems does not create the possibility of a new or different kind of accident.

Therefore, operation of the facility in accordance with the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS change will remove the load limit over the SFSP and CDPS restrictions when the reactor building crane is used with single-failure-proof handling systems that comply with criteria in Section 5.1.6 of NUREG–0612.

The reactor building crane was upgraded to single-failure-proof in compliance with NUREG–0554. The upgraded crane and handling system is in compliance with NUREG–0612, Sections 5.1.1 and 5.1.6. The NRC in NUREG–0612, Section 5.2 documented their review of the potential consequences of a load drop when handled by a single-failure-proof crane using single-failure-proof rigging compared with other alternatives and concluded as follows:

''The likelihood for unacceptable consequences in terms of excessive releases of gap activity or potential for criticality due to accidental dropping of postulated heavy loads after Receptionist (OWFN and TWFN) implementation of the guidelines of Section 5.1 is very low.''

Therefore, there is a very minimal chance of a load drop that could result in consequences that exceed the regulatory accepted standards when the load is handled by a single-failure-proof crane and handling system, and performed in accordance with Section 5.1 of NUREG–0612. A single-failure-proof crane design incorporates the applicable design basis event that in this case is a seismic event. A load drop is of such low probability that it is considered unlikely when it is handled with the reactor building crane because the crane and its handling systems satisfy the NUREG–0612 criteria for a single-failure-proof crane. Therefore, any load lifted over the SFSP using the reactor building crane, and adhering to NUREG–612 Phase I guidelines has a very low probability of falling into the spent fuel pool accidentally or as a result of a design basis event.

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

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

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: September 10, 2001.

Description of amendment request: The proposed change would revise the requirement for the source range monitor (SRM) operability during core operations. The proposed change would require two SRM channels to be operable, one with its detector located in the core quadrant where core alterations are being performed, and another with its detector located in an adjacent quadrant.

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

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

The proposed change revises Technical Specification 3.9.D for source range monitor operability requirements during core alterations. The only accident described in the Final Safety Analysis Report (FSAR) while the plant is in Cold Shutdown or Refueling is a fuel handling (dropped bundle) accident. The proposed change involves equipment that is not involved in the mitigation or prevention of a fuel handling accident as described in FSAR. Therefore, the change to SRM operability requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to TS 3.9.D does not involve any physical alteration of plant equipment or system configuration. Core reactivity and reactivity control functions are not affected, and adequate reactivity monitoring capability is maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change to TS 3.9.D affects the operability requirements for source range monitors during core alterations. The SRMs do not perform any required functions for mitigating the consequences of an accident. The current specification only requires one operable SRM. The proposed specification will ensure redundant monitoring is available to detect changes in the reactivity condition of the core by requiring the operability of at least two source range monitors. This will provide adequate capability for detecting an inadvertent criticality. 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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendment request: September 11, 2001

Description of amendment request: The purpose of the proposed revision to the Technical Specifications (TSs) is to delete the cycle-specific footnote for the Safety Limit Minimum Power Critical Ratio (SLMCPR).

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

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the cycle specific SLMCPR limit for incorporation into the Technical Specification, and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in ''General Electric Standard Application for Reactor Fuel,'' NEDE–24011–P–A–13, and Amendment 25. Amendment 25 was approved by the NRC in a Safety Evaluation Report dated March 11, 1999. The footnote to Technical Specification 2.1.A is being deleted. The footnote associated with the Technical Specification 2.1.A was originally included to ensure that the SLMCPR was only applicable for the identified cycle because Amendment 25 was not yet NRC approved. Amendment 25 has subsequently been approved. Therefore, this footnote is no longer necessary. The footnote was for information only, and has no impact on the design or operation of the plant. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods, which ensures that applicable regulatory requirements are met [. . .]

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

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change deletes the footnote contained in Technical Specification 2.1.A as the result of the NRC approval of Amendment 25 to NEDE–24011–P–A. This change does not affect the design or operation of any plant structures, systems or components. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods, which ensures that applicable regulatory requirements are met. Changes to the SLMCPR value specified in the Technical Specification will require prior NRC approval [. . .]

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

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed change deletes the footnote contained in Technical Specification 2.1.A as the result of the NRC approval of Amendment 25 to NEDE–24011–P–A. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods as specified in the Technical Specifications. These methods ensure that applicable regulatory requirements are met. Changes to the SLMCPR value specified in the Technical Specifications will require prior NRC approval [. . .]

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

NRC Section Chief: L. Raghavan, Acting.

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

Date of amendments request: September 11, 2001.

Description of amendments request: The amendments would allow the non-operating shutdown cooling loop to be declared inoperable for a period up to 2 hours for surveillance testing in MODE 6. The request is based on Technical Specification Task Force Traveler Number 361, Revision 2.

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

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

The proposed amendment would add a note to the limiting condition of operation (LCO) of Technical Specification 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level, that would permit one required SDC loop to be declared inoperable for a period of up to 2 hours for surveillance testing, provided the other SDC loop is OPERABLE and in operation.

Allowing the non-operating SDC loop to be declared inoperable in accordance with the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because the SDC system is not an accident initiator of any previously evaluated accidents. Because the SDC system does not initiate any previously analyzed accidents, it cannot increase the probability of these accidents occurring.

Furthermore, allowing the non-operating SDC loop to be declared inoperable in accordance with the proposed amendment does not involve a significant increase in the consequences of an accident previously analyzed because only one operating SDC loop is necessary to perform the SDC system function of removing decay heat from the reactor core.

The proposed amendment does not represent a change to the design of the facility. Nor does the proposed amendment prevent the safety function of the shutdown cooling system from being performed. The proposed amendment does not alter, degrade, or prevent actions described or assumed in any accident described in the PVNGS Updated Final Safety Analysis Report (UFSAR) from being performed. Therefore, since the SDC system is not an accident initiator and because only one SDC loop is necessary to perform the design function, the proposed amendment would not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed amendment would add a note to the limiting condition of operation (LCO) of Technical Specification 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level, that would permit one required SDC loop to be declared inoperable for a period of up to 2 hours for surveillance testing, provided the other SDC loop is OPERABLE and in operation. Allowing the non-operating SDC loop to be declared inoperable in accordance with the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because: (1) The proposed amendment does not represent a change to the design of the plant, (2) the proposed amendment does not involve the installation of new or different equipment, (3) the proposed amendment does not alter the methods for operating plant equipment, and (4) the proposed amendment does not affect any other safety related equipment. Therefore, the proposed amendment 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 a margin of safety.

The proposed amendment would add a note to the limiting condition of operation (LCO) of Technical Specification (TS) 3.9.5, Shutdown Cooling (SDC) and Coolant Circulation—Low Water Level, that would permit the non-operating SDC loop to be declared inoperable for a period of up to 2 hours for surveillance testing in MODE 6, when the water level is less than 23 feet above the top of the reactor vessel flange, provided the other SDC loop is OPERABLE and in operation. Allowing the non-operating SDC loop to be declared inoperable in accordance with the proposed amendment does not involve a significant reduction in a margin of safety because the operating SDC loop provides sufficient decay heat removal capacity. The proposed change does not impact the operating SDC loop. In the unlikely event that the operating SDC loop becomes inoperable concurrent with the inoperability of the non-operating SDC loop allowed by the proposed note, adequate controls exist within the TS 3.9.5 Required Actions to ensure adequate decay heat removal. In addition, if the operating SDC loop fails, operator action to restore the SDC loop being tested to OPERABLE status and place that SDC loop in operation will be timely such that adequate decay heat removal capability is maintained. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the responses to these three criteria, Arizona Public Service Company (APS) has concluded that the proposed amendment involves no significant hazard consideration.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three 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: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: November 7, 2001.

Description of amendments request: The proposed license amendments would revise Technical Specification (TS) 3.1.4, ''Control Rod Scram Times'' and TS 5.5.10, ''Technical Specifications Bases Control Program.'' TS 3.1.4 would be revised to better delineate the requirements for testing control rod scram times following refueling outages. TS 5.1.10 would be revised to reference Title 10 of the Code of Federal Regulations (10 CFR) Section 50.59. This license amendment application incorporates the NRC-approved Technical Specification Task Force (TSTF) Item 222, Revision 1, ''Control Rod Scram Time Testing,'' and TSTF Item 364, Revision 0, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59.''

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

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

The proposed change to adopt TSTF–222, Revision 1, is an administrative clarification of existing Technical Specification requirements regarding scram time testing requirements for control rods. The current wording of Surveillance Requirement 3.1.4.1 requires each control rod to be tested if any fuel movement occurs in the reactor pressure vessel. Surveillance Requirements 3.1.4.3 and 3.1.4.4 require only the affected control rods to be tested. The NRC-approved TSTF–222, Revision 1, clarifies that post-refueling scram time testing of control rods only applies to control rods affected by work activities. The requirement to test all control rods following routine refueling outages remains unchanged. As such, there is no effect on initiators of analyzed events or assumed mitigation of accidents or transients. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to adopt TSTF–364, Revision 0, is an administrative change to provide consistency between the Technical Specification requirements for the Technical Specification Bases Control Program and the regulatory requirements of Title 10, Section 50.59 of the Code of Federal Regulations, as revised by the NRC on October 4, 1999. The change will have no affect on the initiators of analyzed events or assumed mitigation of accidents or transients.

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

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

The proposed changes to adopt TSTF–222, Revision 1 and TSTF–364, Revision 0, do not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed changes do not create a new or different kind of accident from any accident previously evaluated.

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

The proposed change to adopt TSTF–222, Revision 1, will not reduce a margin of safety because it has no effect on any safety analysis assumptions. The proposed license amendment implements an administrative clarification to better delineate the requirements for scram time testing control rods following refueling outages and for control rods requiring testing due to work activities. The requirement to test all control rods following a routine refueling outage remains unchanged. As such, the proposed change does not involve a significant reduction in the margin of safety.

The proposed change to adopt TSTF–364, Revision 0, is an administrative change to provide consistency between the Technical Specification requirements for the Technical Specification Bases Control Program and the regulatory requirements of Title 10, Section 50.59 of the Code of Federal Regulations, as revised by the NRC on October 4, 1999. The change will not reduce the margin of safety because the change has no effect on any safety analyses assumptions. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: August 24, 2001.

Description of amendment request: The proposed amendment would delete the Technical Specification (TS)-required action which, in the event of inoperability of the oscillation power range monitor (OPRM) trip function, limits plant operation above 25-percent power to 120 days. Instead, continued plant operation would be allowed if a TS-required action is taken to implement an alternate method to detect and suppress thermal-hydraulic instability oscillations.

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

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

The OPRM function is not considered as an initiator of any previously analyzed accident. Therefore, this proposed change does not significantly increase the probability of such accidents. This proposed change would allow the use of existing well-established alternate methods to detect and suppress the thermal hydraulic instability oscillations. Considering that multiple Boiling Water Reactor plants, including Fermi 2, have satisfactorily operated using alternate stability monitoring methods for extended periods of time prior to the installation of OPRM systems, it is concluded that these measures are adequate. Therefore, the consequences of a previously analyzed accident would not be significantly increased.

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

The proposed change does not involve a physical alteration of the plant, add any new equipment, or require any existing equipment to be operated in a manner different from the present design. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed change would allow the use of an existing alternate method to detect and suppress thermal hydraulic instability oscillations to continue to operate the reactor above 25% power in the event of the inoperability of the OPRM system. Considering that multiple Boiling Water Reactor plants, including Fermi 2, have satisfactorily operated using alternate stability monitoring methods for extended periods of time, it is concluded that these measures are adequate, and that the proposed change does not significantly reduce the margin of safety.

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

Attorney for licensee: Peter Marquardt, Legal Department, 688 WCB, Detroit Edison Company, 2000 2nd Avenue, Detroit, Michigan 48226–1279.

NRC Section Chief: William D. Reckley.

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

Date of amendment request: October 23, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) surveillance requirement (SR) 3.8.4.1 to change limits for the battery terminal voltage when on a float charge for 125 VDC station battery 31 following the replacement of this battery in early 2002. The proposed amendment would also revise the applicable TS Bases section.

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 TS SR change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The newly installed battery 31 will consist of 59 cells, instead of the presently installed 58-cell battery. An additional cell will be added to 31 Battery in order to provide an acceptable design margin for future load addition to this battery.

The resulting change in the minimum 31 Battery terminal voltage on float charge to 125.7 V is due to the additional cell added. This new value will ensure that the 31 Battery is properly verified to be functional to meet its design requirements. Calculations demonstrated in IP3–ECCF–845 indicate that 31 Battery DC circuit coordination is not affected by the proposed replacement of the existing battery with a 59-cell battery. The proposed TS SR change does not affect accident initiators or precursors, nor do they alter design assumptions for the systems or components used to mitigate the consequences of an accident as analyzed in Chapter 14 of the IP3 UFSAR [Indian Point 3 Updated Final Safety Analysis Report].

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

No. This TS SR change for 31 Battery is based upon replacement of the 31 Battery with a new 59-cell battery. This new battery 31 is at least equivalent to the existing 58-cell 31 Battery. This new 31 battery, with the added cell, provides an acceptable design margin to the 31 Battery. Battery 31 circuit coordination is not adversely affected by the addition of this new battery with 59 cells. The proposed changes to this TS SR do not introduce any new accident initiators or precursors, or any new design assumptions for those components used to mitigate the consequences of an accident.

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

No. During the replacement of the existing 31 battery with a new 59-cell battery and the subsequent TS SR change that verifies higher minimum terminal voltage on float charge, the new 31 battery and the requirements associated with verifying its design functionality will not involve a significant reduction in the margin of safety. The replacement 31 Battery is at least equivalent to the existing battery. The additional cell in the proposed new 59-cell battery provides an acceptable design margin, which will be 120% for 31 battery with 59 cells. The increase in the number of cells from 58 to 59 will result in a higher 31 Battery terminal voltage on float charge. This proposed TS SR simply documents the verification of this new minimum voltage value. The minimum terminal voltage value for the new 32 Battery will not change nor be impacted by this TS change. Accordingly, there is no significant reduction in [a] margin of safety.

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

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

NRC Section Chief: L. Raghavan (Acting).

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

Date of amendment request: October 15, 2001.

Description of amendment request: The proposed change to Technical Specification 3.4.7 limits Reactor Coolant System activity permitted by the ACTION statement to 60 microcuries per gram (μCi/gm) at all power levels. The letdown line break accident analysis in the Final Safety Analysis Report is also changed to reflect revised dose consequences.

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

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

Response:

The proposed change to the Technical Specifications (TS) conservatively limits Reactor Coolant System (RCS) activity permitted by Action Statement 3.4.7.a to 60 μCi/gm at all reactor power levels. The proposed change to the Final Safety Analysis Report (FSAR) Section 15.6.3.1 revises the letdown line break accident analyses.

The probability of a previously evaluated accident is not affected by this change because the pre-existing iodine spike is not an accident initiator and the FSAR change does not affect any plant Structure, Systems, or Component (SSC) but merely determines the consequences of the previously evaluated accident.

This TS change is conservative in that it will reduce the accident consequences for events occurring at lower power levels.

The proposed FSAR change meets the original SER [Safety Evaluation Report] acceptance criteria with the exception of the Exclusion Area Boundary (EAB) accident induced iodine spiking thyroid dose. The SRP [Standard Review Plan] acceptance criteria for the EAB accident induced iodine spiking thyroid dose is a small fraction of the 10 CFR [Part] 100 limits (30 rem). The proposed change falls well within 10 CFR [Part] 100 limits (75 rem).

The EAB accident induced iodine spiking thyroid dose consequences are considered acceptable and reasonable for the following reasons:

• The letdown line break event starting from the most limiting parameters allowed by the TS LCO [Limiting Conditions for Operation] on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560 °F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.

• Currently, FSAR Table 15.6–4 lists the 'Realistic' EAB thyroid dose as 0.46 rem. The realistic dose is based upon no iodine spike, 50 percentile X/Q [atmospheric dispersion factor], and 0.12% failed fuel RCS activity. The best estimate dose consequences using the new analysis methodology with the normal plant operating parameters would remain below 0.46 rem even for the accident induced iodine spiking event.

• The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors.

The letdown line break consequences are considered acceptable due to the unlikeliness of the event and conservative nature of the analyses. The 'no iodine spike' results remain within a small fraction of the 10 CFR [Part] 100 limits; the 'accident induced iodine spike' results fall well within the 10 CFR [Part] 100 limits; and the 'pre-existing iodine spike' results are within the 10 CFR [Part] 100 limits.

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

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

Response:

The probability of a new or different accident is not affected by this change because the pre-existing iodine spike is not an accident initiator and the FSAR change does not affect any plant Structure, Systems, or Components (SSC) but merely determines the consequences of the previously evaluated accident.

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

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

Response:

The TS change is more limiting in that it will reduce the accident consequences for events occurring at lower plant levels.

The proposed FSAR change meets the original SRP acceptance criteria with the exception of the Exclusion Area Boundary (EAB) accident induced iodine spiking thyroid dose. The SRP acceptance criteria for the EAB accident induced iodine spiking thyroid dose is a small fraction of the 10 CFR [Part] 100 limits (30 rem). The proposed change falls well within 10 CFR [Part] 100 limits (75 rem).

The EAB accident induced iodine spiking thyroid dose consequences are considered not to be a significant reduction in the margin of safety for the following reasons.

• The letdown line break event starting from the TS LCO on RCS activity, pressure, temperature, primary to secondary leakage, and proceeding unmitigated for 30 minutes is highly unlikely. The additional use of conservative assumptions such as an iodine spiking factor of 500, maximum bounding letdown flow, worst case 95 percentile atmospheric dispersion factors, flashing fraction based on 560 °F even though the break flow would travel through the regenerative heat exchanger and cool down, no activity plate out, no ground deposition, and no activity decay in the transit to the exclusion area boundary significantly increases the overall conservative nature of the calculation.

• The FSAR Table 15.6–4 lists the ''Realistic'' EAB thyroid dose as 0.46 rem. The realistic dose is based upon no iodine spike, 50 percentile X/Q, and 0.12% failed fuel RCS activity. The best estimate dose consequences using the new analysis methodology with the normal plant operating parameters would remain below 0.46 rem even for the accident induced iodine spiking event.

• The new analysis accident induced iodine spiking results would remain below the SRP acceptance criteria if any one of the following normal plant operating parameters were used: RCS steady state activity, iodine spiking factor, letdown flow, or atmospheric dispersion factors.

The letdown line break consequences are considered acceptable due to the unlikeliness of the event and conservative nature of the analyses. The ''no iodine spike'' results remain within a small fraction of the 10 CFR [Part] 100 limits; the ''accident induced iodine spike'' results fall well within the 10 CFR [Part] 100 limits; and the ''pre-existing iodine spike'' results are within the 10 CFR [Part] 100 limits.

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: N. S. Reynolds, Esquire, Winston & Strawn 1400 L Street NW., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

[Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois]

Date of amendment request: September 21, 2001.

Description of amendment request: The proposed amendment would revise the Reactor Core Safety Limit (SL) for peak fuel centerline temperature from less than or equal to 4700 °F (i.e., the current TS limit) to the design basis fuel centerline melt temperature of less than 5080 °F, for unirradiated fuel, decreasing by 58 °F per 10,000 Megawatt-Days per Metric Tonne Uranium (MWD/MTU) burnup.

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

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

The use of high burnup rods or assemblies will not increase the probability of any accident previously evaluated. These high burnup rods or assemblies will continue to satisfy all fuel mechanical, nuclear, thermal-hydraulic, and transient analysis design criteria.

Fuel type is not directly related to the probability of any previously evaluated accidents; however, adhering to applicable design criteria and standards precludes challenges to components and systems that could increase the probability of an accident. The high burnup fuel rods will continue to satisfy the Specified Acceptable Fuel Design Limits (SAFDLs) specified in the Westinghouse Topical Report, WCAP–12488–A, ''Westinghouse Fuel Criteria Evaluation Process,'' which was approved by the Nuclear Regulatory Commission (NRC) on July 27, 1994. The clad integrity of the four high burnup rods in the LTA will be maintained as the LTAs will be placed in non-limiting core locations as permitted by TS 4.2.1 and will continue to meet the safety parameter requirements. In addition, the acceptability of using the four high burnup rods in an LTA is evaluated in the Byron Station, Unit 2, Cycle 10 Reload Safety Evaluation which is supported by Westinghouse Topical Report, ''Extended Burnup Operation Assessment for the VANTAGE+ Design in Byron, Unit 2, Cycle 10,'' dated March 2001.

It has been shown in Westinghouse Topical Report, WCAP–12610–P–A, ''VANTAGE+ Fuel Assembly Reference Core Report,'' approved by the NRC in April 1995, that even though there are variations in core inventories of isotopes due to extended burnup up to 75,000 MWD/MTU, there are no significant increases of isotopes that are major contributors to accident doses. It is worthy to note that, at higher burnups, there is a reduction in certain isotopes that are major dose contributors under accident situations (e.g., Kr–88). With only four high burnup rods in the entire core, any variation of isotopes will be extremely small. Thus, the radiation dose limitations of 10 CFR [Part] 100, ''Reactor Site Criteria,'' will not be exceeded.

The bases for establishing the fuel centerline melt temperature are discussed in WCAP–12610–P–A, noted above, and implemented by Westinghouse Topical Report WCAP–14483–A, ''Generic Methodology for Expanded Core Operating Limits Report,'' approved by the NRC on January 19, 1999. These methodologies and associated analyses confirm that the present analytical limits for all accidents will be maintained.

Based on this evaluation, it is concluded that the proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

As required by WCAP–12488–A, the LTA with the four high burnup rods must satisfy the five guidelines accepted by the NRC. These guidelines are as follows:

- Design of LTAs are mechanically and hydraulically compatible with existing fuel
- Peaking factors meet the TS limits
- NRC approved/accepted safety/design methods and codes are used
- No SAFDLs are exceeded
- Not more than eight LTAs per core are inserted

As previously noted, TS 4.2.1 allows the use of a limited number of LTAs in nonlimiting core regions.

The use of high burnup rods or assemblies will comply with WCAP–12488–A and TSs. All safety evaluations in support of using high burnup rods or assemblies have been performed in accordance with accepted methodologies.

In support of proposed High Burnup LTA Programs in the industry, the NRC has requested fuel characterization inspections prior to high burnup irradiation. LTA M09E, (i.e., the assembly containing the high burnup fuel rods at Byron Station) was subjected to fuel characterization inspections prior to operation in Byron Station, Unit 2, Cycle 10. These inspections included assembly growth, rod growth, assembly bow, peripheral rod oxidation, grid growth, grid oxidation, guide thimble inner diameter oxidation, grid cell size, crud scraping, single rod exams for the high burnup rods, profilometry, and pellet-to-pellet gap measurements using a Gamma Scanner instrument. All parameters inspected were found to be acceptable.

By performing the above inspection regimen, the demonstrated adherence to the inspection standards and acceptance criteria precludes the potential for new risks to components and systems that could introduce a new type of accident.

Based on this evaluation, 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?

There is no significant reduction in the margin of safety due to the proposed change. The current TS Safety Limit (SL) 2.1.1.3 states that ''In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700 °F.'' The TS Safety Limit Bases states that overheating of the fuel is prevented by maintaining the steady state peak Linear Heat Rate (LHR) below the level at which fuel centerline melting occurs. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of fuel is high enough to cause the fuel centerline temperature to reach the fuel melting point.

WCAP–14483–A conservatively states that the fuel centerline temperature limit has been established based on the melting temperature for Uranium Dioxide (UO_2) fuel of 5080 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup. Based on the WCAP–14483–A equation, a burnup of approximately 65,500 MWD/MTU could be accrued before the melting temperature would academically reach the current TS SL of 4700 °F.

Westinghouse has evaluated the fuel centerline temperatures for the Byron Station and Braidwood Station reactor cores under uprated power conditions. This evaluation shows that the high burnup rods' temperatures would remain below both the current SL of 4700 °F and the proposed WCAP–14483–A equation (i.e., the proposed SL) for fuel melting temperatures under extended burnup conditions past 75,000 MWD/MTU. Thus, fuel melting will not occur in the LTA high burnup rods.

The insertion of the four high burnup rods does not impact any other TS. The LTA has been designed to operate within the SAFDLs and will therefore have sufficient safety margins. Furthermore, the high burnup LTA will satisfy the five guidelines specified in WCAP–12488–A approved by the NRC. The high burnup LTA will comply with TS 4.2.1 by being placed in a nonlimiting core region.

Based on the above discussion, changing the fuel centerline melt temperature from the existing 4700 °F to an equation consistent with the design basis for fuel melt temperature will not significantly reduce the margin of safety. The analysis shown in WCAP–12610–P–A indicates that the minimum margin to safety occurs at fuel assembly Beginning of Life (BOL). The evaluation in WCAP–12610–P–A demonstrates that margin of safety with respect to the proposed SL equation remains sufficient for fuel burnups up to 75,000 MWD/MTU.

Based on this evaluation, the proposed TS changes do not involve a significant reduction in a margin of safety.

Conclusion: Based upon the above analyses and evaluations, we have concluded that the proposed change to the TS involves no significant hazards consideration.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: September 14, 2001.

Description of amendment request: Exelon proposed to extend the use of the pressure temperature limits specified in Technical Specification (TS) Figure 3.4.6.1–1, ''Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure,'' through Cycle 10 of operation, currently scheduled to end April 2004. Exelon also proposed to modify TS Table 4.4.6.1.3–1, ''Reactor Vessel Material Surveillance Program—Withdrawal Schedule,'' with a note clarifying that surveillance capsule withdrawals are to be scheduled for the nearest vessel refueling outage date subsequent to the withdrawal time specified in the TS Table.

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

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

Extended Use of Pressure-Temperature Limits

The proposed change to the TSs to extend the use of the P–T limits does not affect the operation or configuration of any plant equipment. Thus, no new accident initiators are created by this change. The proposed change extends the use of the pressure-temperature (P–T) limits for an additional cycle of operation. The P–T curves prohibit operational conditions in which brittle fracture of the reactor vessel materials is possible. The P–T limits are based on the projected reactor vessel neutron fluence at 32 effective full power years (EFPY) of operation. At the end of the next cycle of operation, Cycle 10, Limerick Generating Station (LGS) Unit 1 will have attained a maximum of 48.1 percent of the 32 EFPY operating time which provides significant margin to ensure that the current 32 EFPY fluence projection will not be exceeded. This ensures that the basis for proposed applicability of the P–T limits is conservative and that the reactor vessel integrity is protected under all operating conditions. Therefore, neither the probability nor the consequences of an accident are increased.

Deferral of Withdrawal of Vessel Surveillance Specimens

Deferring the withdrawal of the vessel surveillance capsules will not initiate or is not a precursor to any of the accident scenarios presented in the Updated Final Safety Analysis Report (UFSAR). This schedular adjustment will not increase the likelihood of equipment failure, will not defeat the design reactor protection functions, and will not increase the likelihood of failure of any plant structure, system or component. Therefore, neither the probability nor the consequences of an accident are increased.

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

Extended Use of Pressure-Temperature Limits

The proposed change to the technical specifications to extend the use of the P–T limits does not affect the operation or configuration of any plant equipment. The current P–T limits will remain valid and conservative during the proposed extension. Thus, no new or different accidents are created by this proposed change.

Deferral of Withdrawal of Vessel Surveillance Specimens

The proposed deferral of the withdrawal of the vessel surveillance capsule does not involve a change to the plant design or operation. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. Because the P–T limit curves are not impacted, the safety function of the reactor vessel to mitigate the release of radioactive steam and limit reactor inventory loss under normal, accident, and transient conditions is not affected. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Extended Use of Pressure-Temperature Limits

The proposed change extends the use of the current P–T limits for an additional cycle of operation. No changes to the P–T limits are proposed. The current P–T limit curves are based on the projected reactor vessel neutron fluence after 32 EFPY of operation. At the end of the next operating cycle, Cycle 10, LGS Unit 1 will have attained a maximum of 48.1 percent of the 32 EFPY reactor vessel neutron fluence projection upon which the current P–T curves are based. The maximum operating time at the end of Cycle 10, when compared with the maximum operating time assumed for the P–T limits curves, ensures that the P–T limits will remain conservative and will ensure that the current margins for reactor pressure vessel integrity are unchanged. The proposed change maintains the relative margin of safety commensurate with that which existed at the time the American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section XI, Appendix G, was approved in 1974. No plant safety limits, setpoints, or design parameters are adversely affected by the proposed TS change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Deferral of Withdrawal of Vessel Surveillance Specimens

No plant safety limits, set points, or design parameters are adversely affected by the proposed deferral of withdrawal of vessel surveillance specimens. The deferral of the withdrawal of the vessel surveillance specimens does not affect the current P–T limit curves, and therefore, does not affect a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 18, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for St. Lucie, Units 1 and 2, regarding Engineered Safety Feature Actuation System (ESFAS) instrumentation. Specifically, it would limit the period of time that inoperable recirculation actuation signal (RAS), containment spray actuation signal (CSAS), and auxiliary feedwater actuation signal (AFAS) input channels could be in the bypassed and/or tripped condition. Generally, the proposed TS employ a 48-hour completion time to restore an inoperable channel, which, in most cases, is more restrictive than the existing TS, and is comparable to the value used in the Standard TS for Combustion Engineering plants.

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

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

No, facility operation under the new Technical Specification (TS) restrictions would not increase the probability of occurrence of any accident previously evaluated. The proposed changes only affect the ESFAS functions of RAS, CSAS, and AFAS; generally limiting the time that any instrument channel may be inoperable in a bypassed or tripped condition. No physical plant changes are proposed in conjunction with these revisions. The proposed changes to RAS and AFAS channel operability greatly reduce the time that actuation systems are vulnerable to spurious, inadvertent actuation. The proposed changes do allow a new unlimited time for trip of one CSAS channel on Unit 1. Although this increases the possibility of a spurious channel trip with a potential for causing an inadvertent spray actuation, this is offset by the increased reliability of spray in this configuration. Unit 2 already contains provision for the indefinite single channel trip of CSAS, and this change will also make the two units similar. Additionally, it is important to note that inadvertent actuation of any of these functions (RAS, CSAS, or AFAS) during plant operation is not an accident initiating event. Therefore, with no physical effects on the plant and no increase in probability that the subject ESFAS functions will initiate an accident, there is no increased probability that any previously evaluated accident will occur. The changes provided in this safety evaluation do not affect the assumptions or results of any accident evaluated in the UFSAR [Updated Final Safety Analysis Report].

Likewise, the consequences of any accident previously evaluated have not been increased. The proposed changes, by limiting the time that ESFAS functions are inoperable, will increase the reliability of the associated ESFAS functions to respond to accidents. In particular, the revision to the RAS TS will limit the time that the RAS will be vulnerable to single failure and will therefore improve the system reliability during an accident. As these proposed changes constitute no physical change to the facility and only serve to increase ESF function reliability, FPL concludes that the consequences of previously evaluated accidents are not increased. The ability of the ESFAS to respond to accident conditions as assumed in any accident analysis has not been affected.

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

No, the proposed activity does not create the possibility of an accident of a different type than any previously evaluated. The proposed changes only affect the ESFAS functions of RAS, CSAS, and AFAS; generally limiting the time that any instrument channel may be inoperable in a bypassed or tripped condition. No physical plant changes are proposed in conjunction with these revisions. Thereby, the proposed changes do not create any new equipment interfaces, equipment response characteristics, or operating configurations. Without creation of a new interaction of materials, operating configuration, or operating interface, there is no possibility that the proposed changes can introduce a new or different kind of accident.

(3) Would operation of the facility in accordance with the proposed amendments involve a significant reduction in a margin of safety?

The margin of safety as defined in the basis for any Technical Specification or in any licensing document has not been reduced. The TS Bases for the associated ESF LCO [Limiting Condition for Operation] do not explicitly discuss a related margin of safety. However, by virtue of the increased ESFAS reliability provided by the proposed amendments, it is evident that the margin of safety will not be reduced in any manner.

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

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

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of amendment request: October 17, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 6.8.4.h, ''Containment Leakage Rate Testing Program,'' to allow only one-time deviation from the 10-year frequency of the performance-based leakage rate testing program for Type A tests as recommended by Nuclear Energy Institute, NEI 94–01, Revision 0, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR part 50, Appendix J,'' and endorsed by Regulatory Guide 1.163, ''Performance-Based Containment Leak-Rate Program.'' The one-time deviation would allow intergrated leak rate testing (ILRT) at no more than 15 years after the last ILRTs, performed in November 1992 and October 1991 for Units 3 and 4 respectively.

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

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment to the Technical Specifications adds a one-time extension to the current interval for Type A (ILRT) testing. The current test interval of ten years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification, nor a change to the operation of the plant, and the test extension is not a type that could lead to equipment failure or accident initiation. The proposed extension of Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG–1493 has found that, generically, very few potential containment leakage paths are not identified with Type B and C tests. In fact, an analysis of 144 ILRT results, including 23 failures, found that no failures were due to containment liner breech. The NUREG concluded that reducing the Type A frequency to one per twenty years was found to lead to an imperceptible increase in risk.

Florida Power & Light 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 four Type A tests for both Turkey Point Units 3 and 4 show leakage rates well below acceptance criteria, indicating a leak-tight containment. Inspections required by the Maintenance Rule [10 CFR 50.65] and ASME [American Society of Mechanical Engineers] code, will identify indications of containment structure degradation that could affect that leak tightness. Type B and C testing required by Technical Specifications will identify any containment openings, such as valves, that would otherwise be detected by the Type A tests. These factors show that the Turkey Point Units 3 and 4 Type A test extension will not represent a significant increase in the consequences of an accident.

Based on the above, it is concluded that the proposed amendments to extend the Type A test frequency does not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not create a new or different type of accident for Turkey Point because no physical plant changes are being made, and no compensatory measures are imposed that would create a new failure scenario. The proposed change only requests a one-time extension to the current interval for Type A testing. The current test 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.

The proposed extension to Type A testing cannot create the possibility of a new or different type of accident because 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 or affecting the mitigation of an accident.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed license amendment requests 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 15 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 test interval for Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG–1493 found that, generically, the design containment leakage rate contributed about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have minimal effect on this risk, since 95 percent of the potential leakage paths are detected by Type B and C testing. A Turkey Point plant-specific risk calculation is consistent with the generic conclusions identified in NUREG–1493.

Therefore, the proposed changes in this license amendment will not result in a significant reduction in the plant's 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: M.S. Ross, Attorney, Florida Power & Light, P.O. Box 14000, Juno Beach, Florida 33408–0420.

NRC Section Chief: Richard P. Correia.

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

Date of amendment requests: August 7, 2001.

Description of amendment requests: The proposed amendments would create Technical Specification (TS) 3.0.6 and associated bases to allow equipment that was removed from service or declared inoperable to be returned to service under administrative controls solely to perform the testing required to demonstrate its operability or the operability of other equipment. TS 3.0.6 would incorporate the administrative controls currently approved for use as TS 3.0.5 in NUREG–1431, ''Standard Technical Specifications Westinghouse Plants,'' Revision 2, dated April 30, 2001.

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 of occurrence or consequences of an accident previously evaluated?

Probability of Occurrence of an Accident Previously Evaluated

The potential impact of temporarily returning the equipment to service is considered to be insignificant since the equipment will either be expected to be able to perform its required safety function or sufficient redundancy will exist such that the function would still occur if required. This is addressed in Generic Letter (GL) 87–09, ''Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.'' GL 87–09 states, ''It is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed because the vast majority of surveillances do in fact demonstrate that systems or components are operable.'' In addition, returning the equipment to service for testing will promote timely restoration of the equipment. Therefore, the proposed changes do not significantly affect accident initiators or precursors.

The proposed change to create a Bases statement for TS 3.0.6 provides explanatory information regarding the intent of the specification and how it is to be implemented. The proposed Bases change does not alter requirements of the associated TS. Therefore, the effect of the Bases change on accident initiators and precursors of an accident is bounded by the effect of the TS change as described above. The format changes are intended to improve appearance and do not alter any requirements.

Therefore, the proposed changes do not adversely affect any accident initiators or precursors and will not involve a significant increase in the probability of an accident previously evaluated.

Consequences of an Accident Previously Evaluated

The proposed change will allow temporarily returning equipment, that was previously declared inoperable, to service in a state in which it is expected to function to mitigate the consequences of a previously analyzed accident. The proposed change will also permit temporarily restoring inoperable equipment to service in situations where sufficient redundancy would exist for its function to mitigate the consequences of a previously analyzed accident to be performed. This will promote timely restoration of equipment and capabilities to mitigate the consequences of an accident previously analyzed.

The proposed change to include a Bases statement for TS 3.0.6 provides explanatory information regarding the intent of the specification and how it is to be implemented. The proposed Bases change does not alter requirements of the associated TS. Therefore, the effect of the Bases change on offsite dose consequences of an accident previously analyzed is bounded by the effect of the TS change as described above. The format changes are intended to improve appearance and do not alter any requirements.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not significantly increased.

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

The proposed changes do not introduce a new mode of plant operation and do not involve a physical modification to the plant. Operation with the inoperable equipment temporarily restored to service under administrative controls is not considered a new mode of operation since the equipment is not being physically altered. As such, the manner in which it can fail remains the same.

The proposed change to include a Bases statement for TS 3.0.6 provides explanatory information regarding the intent of the specification and how it is to be implemented. The proposed Bases change does not alter requirements of the associated TS. Therefore, the effect of the Bases changes on accident initiators or precursors is bounded by the effect of the associated TS as described above. The format changes are intended to improve appearance and do not alter any 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 change involve a significant reduction in a margin of safety?

The proposed new TS 3.0.6 can be applied to any structures, systems, and components that are governed by the TS. As such, the proposed changes are applicable to every margin of safety imposed by the TS.

The proposed change will allow temporarily returning equipment that was previously declared inoperable to service in a state in which it is expected to function to mitigate the consequences of a previously analyzed accident. The proposed change will also permit temporarily restoring inoperable equipment to service in situations where sufficient redundancy would exist for its function to mitigate the consequences of a previously analyzed accident to be performed. The performance of the testing should confirm the expected capability of the equipment and there is no significant impact on any TS safety setting or setpoint.

There is no margin of safety pertinent to the proposed Bases change. The format changes are intended to improve appearance and do not alter any requirements.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety. In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no significant hazards consideration.

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

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

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

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

Date of amendment request: May 9, 2001.

Description of amendment request: The proposed amendment would change the Technical Specification (TS) to correct an error in TS Table 3.3.1.1–1 Function 2.b, correct a typographical error in labeling surveillance requirement 3.3.1.1.13, and revise bases pages B 3.3–8 and B 3.3–10.

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

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

This change is to correct an error in documentation that was introduced during implementation of Amendment 151 and retained in TS during the conversion to ITS [Improved Technical Specifications] as well as an error that was introduced into TS during the conversion to ITS. Neither the design basis nor the functionality of the instrumentation is being physically changed. The Neutron Monitoring system performs a mitigating function and is not an accident initiating system. The actual mitigating function of the Neutron Monitoring is not changed. Only an implied but non-existent mitigating capability is being removed from TS. This change does not create or modify any accident initiators. Therefore, there is no increase in the probability of an accident previously evaluated.

The APRM [Average Power Range Monitor] system is credited for mitigating the consequences of the Control Rod Drop Accident. The APRM system also provides protection for the reactor to mitigate the consequences of such abnormal operational transients as loss of feedwater heater, pressure regulator failure, or Main Steam Isolation Valve closure. The proposed change will not change the functionality or setpoints for either the APRM Flux-High (Fixed) or the APRM Flux-High (Biased) functions. Additionally, the correction of an incorrect Surveillance Requirement reference does not change how any surveillance is performed. Therefore the consequences of an accident previously evaluated will not be increased.

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

Since this change in the TS does not involve a physical change to the instrumentation, to the setpoints, or to the design or functionality of the circuitry for reactor scram on APRM Flux-High, fixed or flow-biased, the change does not create a possibility of a new or different kind of accident not previously evaluated.

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

The setpoints for the Neutron Flux-High instrumentation are not changed by this proposed TS change. The safety function allowable value setpoint remains at less than or equal to 120% RTP [rated thermal power]. The formula for the APRM Flux-High (flow biased) is not being changed. Since neither of these is being changed, the margin of safety is not reduced.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 6, 2001, as supplemented November 2, 2001.

Description of amendment request: The proposed amendment would revise the Seabrook Station Technical Specifications (TS) Index, TS 1.0, "Definitions," and TS Table 1.2, "Operational Modes," to reflect the improved Standard Technical Specifications for Westinghouse plants.

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

The proposed changes to TS Index, TS 1.0 and TS Table 1.2 are changes that do not change any structures, systems or components (SSCs) thus, the proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility. In addition, the proposed changes do not affect the manner in which the plant responds in normal operation, transient or accident conditions. The proposed changes do not alter or prevent the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). Finally, while these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety.

The proposed changes do not affect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/ public radiation exposures.

Therefore, it is concluded that these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

This [sic] proposed changes to TS Index, TS 1.0 and TS Table 1.2 are changes that do not change the operation or the design basis of any plant system or component during normal or accident conditions. The proposed change incorporates definitions delineated in the improved Standard Technical Specifications (NUREG–1431). The proposed changes do not include any physical changes to the plant. In addition, the proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes.

The proposed changes are administrative in nature and only correct, update and clarify the Seabrook Station Operating License to reflect the definitions in the improved Standard Technical Specifications. The proposed changes do not modify the facility nor do they affect the plant's response to normal, transient or accident conditions. The changes do not introduce a new mode of plant operation. While these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety. The plant's design and design basis are not revised and the current safety analyses remains in effect.

Thus, these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The proposed changes to TS Index, TS 1.0 and TS Table 1.2 are administrative in nature and only correct, update and clarify the Seabrook Station Operating License to reflect the improved Standard Technical Specifications. While these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications are not revised nor is the plant design revised by the proposed changes.

Thus, it is concluded that these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not involve a significant reduction in a margin of safety.

Based on the above evaluation, North Atlantic concludes that the proposed changes to TS Index, TS 1.0 and TS Table 1.2 do not constitute a significant hazard.

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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: November 2, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 3.3.1–1, Item 1, ''Variable High Power Trip'' (VHPT), by increasing the maximum allowable value for the VHPT from 106.5 percent to 111 percent.

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:

Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of Amendment.'' The following evaluation supports the finding that operation of the facility in accordance with the proposed change would not:

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

The proposed change to the maximum Allowable Value for the Variable High Power Trip (VHPT) function in the Technical Specifications would not change or remove any considerations of uncertainties from the FSAR [Final Safety Analysis Report] Chapter 14 Safety Analysis. The methodology that was utilized in determining the recommended change in the maximum allowable value follows standard ANSI/ISA–S67.04–1994, ''Setpoints for Nuclear Safety-Related Instrumentation,'' and NRC Regulatory Guide 1.105, ''Setpoints for Safety-Related Instrumentation,'' Revision 3. With the proposed changes to the maximum allowable value and calculated setpoint of the VHPT in place, the reactor is still protected from reaching the analytical limit of 115% reactor power.

Therefore, operation of the facility in accordance with the proposed change to the Technical Specifications would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes to the maximum Allowable Value and Calculated Setpoint for the Variable High Power Trip function in the Technical Specifications would not change or add a system function. The proposed change alters the way the uncertainties (including uncertainties of instrument measurement and calibration) are accounted for without actually removing uncertainties from the calculation. This proposed change follows the standard ANSI/ISA–S67.04–1994 and NRC Regulatory Guide 1.105, Revision 3.

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

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

The proposed change to the maximum Allowable Value for the Variable High Power Trip function in the Technical Specifications would account for all uncertainties in the VHP trip setpoint calculation, instead of taking them into account in the maximum allowable value calculation, as is currently done. In addition, double accounting for nuclear instrumentation uncertainties has been removed. The uncertainties will still be taken into account in determining the calculated setpoint based on the maximum allowable value of the VHPT, in accordance with the standard ANSI/ISA–S67.04–1994 and NRC Regulatory Guide 1.105, Revision 3. This methodology continues to assure that the Analytical Limit will not be exceeded.

Therefore, the proposed change to the Technical Specifications would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based upon 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: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: William D. Reckley (Acting).

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

Date of amendment request: November 1, 2001.

Description of amendment request: The proposed amendments would change the Technical Specifications (TSs) to allow a one-time extension of the allowed outage time for the control room emergency filtration system (CREFS) from 7 days to 30 days. The licensee is requesting this one-time change in order to implement modifications to the CREFS.

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

1. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.

The operability of CREFS ensures that the control room will remain habitable for operators during and following all credible accident conditions. The inoperability or failure of CREFS is not an accident initiator or precursor. Therefore, the probability of an accident previously evaluated will not be significantly increased as a result of the proposed change. Because design limitations continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. Control room dose calculations are not affected outside the limited one-time period when the CREFS modifications/upgrades are ongoing.

During the period that CREFS will be inoperable, temporary ventilation will provide adequate filtration to the control room and adequate cooling to the control and computer rooms. The effectiveness of the temporary filtration provided during this 30 day period is not significantly less than that of the permanently installed CREFS. Only the duration of a currently allowed outage time is being changed, with commensurate compensatory measures being taken. Therefore, the consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

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

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. The evaluation of the effects of the proposed changes indicate that all design standards and applicable safety criteria limits are met. These changes therefore do not cause the initiation of any new or different accident nor create any new failure mechanisms.

Equipment important to safety will continue to operate as designed. Only the duration of a system's allowed outage time is being changed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

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

The CREFS functions to mitigate the effects of accidents. Implementation of the modifications/upgrades will require removing the system from service for a period of time longer than presently allowed by the Technical Specification. This results in a longer period during which the consequences of a design basis accident, affecting the dose of control room personnel, may be slightly increased. During the period that CREFS will be inoperable, a temporary ventilation system will provide adequate filtration to the control room and adequate cooling to the control and computer rooms. The effectiveness of the temporary filtration provided during this 30 day period is not significantly less than that of the permanently installed CREFS. Only the duration of a currently allowed outage time is being changed, with commensurate compensatory measures being taken. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed modification will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other SSCs important to safety. The analysis for the limiting design basis accident, the large break LOCA, has a significant amount of conservatism built in to account for uncertainties in system performance an analysis techniques. This conservative margin of safety, along with the temporary filtration unit, provide a high level of confidence that the health and safety of the operators will be maintained, such that they will be able to prevent or mitigate an event. Therefore, removing the CREFS from service for 30 days on a one-time basis to permit system upgrading, will not significantly reduce the margin of safety. The improvements to CREFS resulting from the proposed modifications will enhance operator protection against conditions resulting from a design basis accident and therefore provide a net benefit to radiological health and reactor safety.

Conclusion

Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and, does not result in a significant reduction in any margin of safety. Therefore, operation of PBNP [Point Beach Nuclear Plant] in accordance with the proposed amendments does not result in a significant hazards determination.

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

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

NRC Section Chief: William Reckley (Acting).

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

Date of amendment request: October 18, 2001.

Description of amendment request: The proposed amendments would modify the Technical Specification Surveillance Requirement (SR) 3.4.3.1 for testing of the main steam safety relief valves (MSRVs) so that the setpoint tolerance for ''As-Found'' testing would be changed from ±1 percent to ± 3 percent. The requirements for testing of the tolerances associated with ''As-left'' testing would remain unchanged. An editorial change would also be made to remove a note regarding an associated relief request.

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 action does not involve a significant increase in the probability or consequences of an accident as previously evaluated.

The proposed change allows an increase in the as-found MSRV safety mode setpoint tolerance, determined by test after the valves have been removed from service, from ±1% to ±3%. The proposed change does not alter the TS 3.4.3 Surveillance Requirements on the nominal MSRV safety mode lift setpoints, the MSRV relief mode setpoints, the required frequency for the MSRV lift setpoint tests, or the number of MSRVs required to be operable.

Consistent with current requirements, this change continues to require that these valves be adjusted to within ±1% of their nominal lift setpoints following testing. The proposed action does not change any other behavior or operation of any MSRV, and therefore, has no significant impact on the reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Final Safety Analysis Report (FSAR).

The proposed action does not involve physical changes to the valves, nor does it change the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components. Therefore, these changes will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC–31753P, ''BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,'' and were reviewed and approved by the NRC in a Safety Evaluation Report (SER) dated March 8, 1993. The plant specific evaluations, required by the NRC's SER and performed to support this proposed change, show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a ±3% setpoint tolerance. These analyses also show that operation of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems are not adversely affected and the containment response from a loss of coolant accident is acceptable. The plant systems associated with these proposed changes are capable of meeting all applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the FSAR. Therefore, these changes do not involve an increase in the consequences of any accident previously evaluated.

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

2. The proposed action does not create a possibility of a new or different kind of accident than previously evaluated.

The proposed change was developed in accordance with the provisions contained in the NRC SER, dated March 8, 1993, for the ''BWR Owners Group Inservice Pressure Relief Technical Specification Revision Licensing Topical Report,'' NEDC–31753P. The revised MSRV setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Since the proposed action does not involve hardware changes, significant changes to the operation of any systems or components, nor changes to existing structures, systems, or components, there is no possibility that a new or different kind of accident is created.

The proposed change to allow an increase in the MSRV safety mode setpoint tolerance from ±1% to ±3% does not alter the nominal MSRV lift setpoints or the number of MSRVs currently required to be operable by SSES Technical Specifications. The proposed action does not involve physical changes to the valves, nor does it change the safety function of the valves. The proposed action does not involve a physical alteration of any existing plant equipment. No new or different equipment is being installed. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the procedures governing normal plant operation, nor the procedures utilized to respond to plant transients.

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

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

The proposed action does not involve a significant reduction in a margin of safety. Establishment of the ±3% MSRV safety setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Engineering evaluations concluded that there are no significant impacts on fuel thermal limits, safety related systems, structures or components, and no significant impact on the accident analyses associated with the proposed changes.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

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: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101–1179.

NRC Section Chief: Lakshminaras Raghaven.

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

Date of amendment request: September 20, 2001.

Description of amendment request: The proposed amendments would support extension of the operating cycle from 18 months to 24 months.

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

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

a. Surveillance Testing Interval Extensions.

The proposed Technical Specification (TS) change involves a change in the surveillance testing intervals to facilitate a change in the operating cycle from 18 months to 24 months. The proposed TS change does not physically impact the plant, nor does it impact any design or functional requirements of the associated systems. That is, the proposed TS change neither degrades the performance of, nor increases the challenges to, any safety systems assumed to function in the plant safety analysis. The proposed TS change neither impacts the TS SRs [surveillance requirements] themselves nor the manner in which the surveillances are performed.

In addition, the proposed TS change does not introduce any accident initiators, since no accidents previously evaluated relate to the frequency of surveillance testing. Also, evaluation of the proposed TS change demonstrates that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident is not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, or the high reliability of the equipment. Since the impact on the systems is minimal, it is concluded the overall impact on the safety analysis is negligible.

Furthermore, an historical review of surveillance test results and associated maintenance records indicate there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not significantly increase the probability or consequences of an accident previously evaluated.

b. Allowable Value Changes.

A change in Allowable Values is proposed for Table 3.3.5.1–1, Item 2.f. The proposed change is the result of application for the Hatch Instrument Setpoint Methodology using plant-specific drift values. Application of this methodology results in Allowable Values that more accurately reflect total instrumentation loop accuracy, as well as that of test equipment and calculated drift between surveillances. The proposed change will not result in any hardware changes. The instrumentation is not assumed to be an initiator of any analyzed event. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to the proposed changes. The role of the instrumentation is in mitigating and thereby, limiting the consequences of accidents.

The Allowable Values were developed to ensure the design and safety analysis limits are satisfied. The methodology used for the development of the Allowable Values ensures: 1) the affected instrumentation remains capable of mitigating design basis events as described in the safety analysis and 2) the results and radiological consequences described in the safety analysis remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

c. Surveillance Testing Interval Reduction to Semiannual.

The proposed TS change involves a reduction in the surveillance testing interval from 18 months to 184 days for the instrumentation associated with Table 3.3.8.2–1. The shorter intervals are based upon the plant-specific results of a review of the surveillance test history for the devices. The implementing procedures for these SRs have been performed on a 184-day interval for a number of years, and this change more accurately reflects actual plant maintenance practices. The proposed, more restrictive TS change does not physically impact the plant, nor does it impact any design or functional requirements of the associated systems. That is, the proposed TS change neither degrades the performance of, nor increases the challenges to, any safety systems assumed to function in the safety analysis. This proposed TS change neither impacts the TS SRs themselves nor the manner in which the surveillances are performed.

In addition, the proposed TS change does not introduce any accident initiators, since no accidents previously evaluated relate to the frequency of surveillance testing. The proposed TS intervals demonstrate that the equipment and systems required to prevent or mitigate the radiological consequences of an accident are continuing to meet the assumptions of the setpoint evaluation on a more frequent basis. Since the impact on the systems is minimal, and the assumptions of the safety analyses are maintained, it is concluded the overall impact on the plant safety analysis is negligible.

Furthermore, setpoint drift evaluations prepared for the subject instrumentation show that the existing Allowable Values are acceptable without change. Therefore, the proposed TS change does not significantly increase the probability or consequences of an accident previously evaluated.

d. Change of CHANNEL CALIBRATION to CHANNEL FUNCTIONAL TEST for Float Switches.

The proposed TS change involves a change in the SRs from CHANNEL CALIBRATIONS to CHANNEL FUNCTIONAL TESTS for float switches used in Table 3.3.1.1–1, Item 7.b; Table 3.3.5.1–1, Item 3.d; and Table 3.3.5.2–1, Items 3 and 4. The float switches are mechanical devices that require mechanical setting at the proper level only. Because the devices cannot be significantly adjusted without a physical change in the location of the installation, the CHANNEL FUNCTIONAL TEST provides all the functionality of a CHANNEL CALIBRATION for this type of device. Therefore, the change in type of SR does not impact the actual testing requirements for the subject devices.

The proposed TS change does not physically impact the plant, nor does it impact any design or functional requirements of the associated systems. That is, the proposed TS change neither degrades the performance of, nor increases the challenges to, any safety systems assumed to function in the safety analysis. The proposed TS change does not impact the manner in which the surveillances are performed.

In addition, the proposed TS change does not introduce any accident initiators, since the same functional requirements exist with the proposed change. Also, evaluation of the proposed TS change demonstrates the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident is not significantly affected because of the availability of redundant systems and equipment and the high reliability of the equipment. Since the impact on the systems is minimal, it is concluded the overall impact on the plant safety analysis is negligible.

Furthermore, an historical review of surveillance test results and associated maintenance records indicated that there was no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed TS change does not significantly increase the probability or consequences of an accident previously evaluated.

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

a. Surveillance Testing Interval Extensions.

The proposed TS change involves a change in the surveillance testing intervals to facilitate a change in the operating cycle length. The proposed TS change does not introduce any failure mechanisms of a different type than those previously evaluated, since there are no physical changes being made to the facility. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure modes are introduced. In addition, the SRs themselves, and the manner in which surveillance tests are performed, remain unchanged.

Furthermore, an historical review of surveillance test results and associated maintenance records indicate there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

b. Allowable Value Changes.

The proposed change in Allowable Values is the result of application of the Instrument Setpoint Methodology using plant-specific drift values and does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based upon the fact that the method and manner of plant operation are unchanged.

The use of the proposed Allowable Values does not impact safe operation of the plant in that the safety analysis limits are maintained. The proposed change in Allowable Values involves no system additions or physical modifications to plant systems. The Allowable Values are revised to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation, except that setpoints may be changed. Since operational methods remain unchanged and the operating parameters were evaluated to maintain the plant within existing design basis criteria, no different type of failure or accident is created.

c. Surveillance Testing Interval Reductions to Semiannual.

The proposed TS change involves a change in the surveillance testing interval due to the review of the surveillance test history of the subject devices. Also, the semiannual tests reflect current HNP calibration practices. The proposed TS change does not introduce any failure mechanism of a different type than those previously evaluated, since the proposed change makes no physical changes to the plant. No new or different equipment is being installed. No installed equipment is being operated in a different manner.

Furthermore, an historical review of surveillance test results and associated maintenance records indicate there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

d. Change of CHANNEL CALIBRATION to CHANNEL FUNCTIONAL TEST for Float Switches.

The proposed TS change does not impact the actual testing requirements for the subject devices. The proposed TS change does not introduce any failure mechanism of a different type than those previously evaluated, since the proposed change makes no physical changes to the plant. No new or different equipment is being installed. No installed equipment is being operated in a different manner. As a result, no new failure mode is being introduced. In addition, the SRs themselves, and the manner in which surveillance tests are performed, remain unchanged.

Furthermore, an historical review of surveillance test results and associated maintenance records indicates there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

a. Surveillance Testing Interval Extensions.

Although the proposed TS change results in changes in the interval between surveillance tests, the impact, if any, on system availability is minimal, based upon other, more frequent testing that is performed, the existence of redundant systems and equipment, or overall system reliability. Evaluations show there is no evidence of any time-dependent failure that would impact the system availability.

The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

b. Allowable Value Changes.

The proposed change does not involve a reduction in a margin of safety. The proposed change was developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in a margin of safety.

c. Surveillance Testing Interval Reductions to Semiannual.

The proposed TS change results in a shorter interval between surveillance tests to ensure the assumptions of the safety analysis are maintained. The impact, if any, on system availability is minimal, as a result of the more frequent testing that is performed. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

d. Change of CHANNEL CALIBRATION to CHANNEL FUNCTIONAL TEST for Float Switches.

The proposed TS change does not impact the actual testing requirements for the subject devices. The impact, if any, on system availability due to this change is minimal, based upon the existence of redundant systems and equipment and overall system reliability.

An historical review of surveillance test results and associated maintenance records indicates there is no evidence of any failure that would invalidate the above conclusions. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: September 20, 2001.

Description of amendment request: The proposed amendments would change specified surveillances from 92 days to 184 days.

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

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

The proposed Technical Specifications (TS) change involves an increase in the surveillance testing intervals for various Surveillance Requirements (SRs) from 92 days to 184 days. The proposed TS changes do not physically impact the plant, nor do they impact any design or functional requirements of the associated systems. That is, the proposed TS change does not degrade the performance of, or increase the challenges to, any safety systems assumed to function in the safety analysis. The proposed TS changes neither impact the TS SRs themselves nor the way in which the surveillances are performed. In addition, the proposed TS change does not introduce any accident initiators, since no accidents previously evaluated relate to the frequency of surveillance testing. Also, evaluation of the proposed TS change demonstrates that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident are not significantly affected because of other, more frequent testing that is performed, the availability of redundant systems and equipment, or the high reliability of the equipment. Since the impact on the systems is minimal, it is concluded that the overall impact on the plant safety analysis is negligible.

A sensitivity analysis was performed to determine the effect of the increased surveillance intervals on the HNP [Hatch Nuclear Plant] Probabilistic Risk Assessment (PRA). This sensitivity analysis shows a negligible increase in core damage frequency (CDF) and essentially no change in large early release frequency (LERF) due to the proposed change.

Furthermore, an historical review of surveillance test results and associated maintenance record indicates there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed TS change involves a change in the various SR intervals from 92 days to 184 days. The proposed TS change does not introduce any failure mechanisms of a different type than those previously evaluated, since no physical changes to the plant are being made. Also, no new or different equipment is being installed, and no installed equipment is being operated in a different manner. As a result, no new failure modes are introduced. In addition, the surveillance test requirements themselves, and the way surveillance tests are performed, remain unchanged.

Furthermore, an historical review of surveillance test results and associated maintenance records indicates there is no evidence of any failure that would invalidate the above conclusions. Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Although the proposed TS change results in changes to the interval between surveillance tests, the impact, if any, on system availability is minimal, based upon other, more frequent testing that is performed, the existence of redundant systems and equipment, or overall system reliability. Evaluations show there is no evidence of time-dependent failures that would impact the availability of the systems. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

A sensitivity analysis was performed to determine the effect of the increased surveillance intervals on the HNP PRA. This sensitivity analysis shows a negligible increase in CDF and essentially no change in LERF due to the proposed change.

Furthermore, an historical review of surveillance test results and associated maintenance records indicates there was no evidence of any failure that would invalidate the above conclusions. 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: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer, Acting.

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

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

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

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

Date of amendment request: October 16, 2001.

Brief description of amendment request: The proposed amendment would add a condition to the Operating License to extend certain Technical Specification surveillance requirement (SR) intervals, one time. The SR intervals would be extended up to 65 days, but no later than April 30, 2003, to permit them to be performed during the next refueling outage, which has been rescheduled because the plant is currently in a forced extended outage.

*Date of publication of individual notice in **Federal Register:*** November 13, 2001 (66 FR 56865).

Expiration date of individual notice: December 13, 2001.

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/NRC/ADAMS/index.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by email to pdr@nrc.gov.

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

Date of application for amendment: December 29, 2000, as supplemented March 22 and July 27, 2001.

Brief description of amendment: The amendment increases the allowed outage time from 3 to 14 days for a single inoperable Division 1 or 2 diesel generator.

Date of issuance: November 8, 2001.

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

Amendment No.: 141.

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

*Date of initial notice in **Federal Register:*** January 24, 2001 (66 FR 7668). 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 November 8, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 2, 2001, as supplemented July 18, 2001.

Brief description of amendment: The amendment extends the surveillance test interval of the slave relays of the Engineered Safety Features Actuation System from 90 days to 8 months.

Date of issuance: November 5, 2001.

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

Amendment No.: 198.

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

*Date of initial notice in **Federal Register:*** July 11, 2001 (66 FR 36337).

The July 18, 2001, supplement was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

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

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: May 2, 2001, as supplemented by letter dated August 23, 2001.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to not require the moderator temperature coefficient (MTC) determination in TS 4.1.1.4.2c if the results of the MTC determination required in TSs 4.1.1.4.2a and 4.1.1.4.2b are within a certain tolerance of the corresponding design values.

Date of issuance: November 16, 2001.

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

Amendment No.: 236.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31706).

The August 23, 2001, supplemental letter provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket No. 50–237, Dresden Nuclear Power Station, Unit 2, Grundy County, Illinois

Date of application for amendment: June 6, 2001, as supplemented by letter dated September 17, 2001.

Brief description of amendment: The amendment revises the values of the Safety Limit for the Minimum Critical Power Ratio in Technical Specification Section 2.1.1.

Date of issuance: November 2, 2001.

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

Amendment No.: 189.

Facility Operating License No. DPR–19: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** September 5, 2001 (66 FR 46479).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 2, 2001.

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: September 29, 2000, as supplemented by letters dated March 1, July 13, August 9, August 13, and October 17, 2001

*Brief description of amendments:*The amendments change the technical specifications to reflect a change in fuel vendors from Siemens Power Corporation to General Electric, and a transition to GE14 fuel. As part of the transition, changes are made to the number of required automatic depressurization system valves and to the time delay relay settings on emergency core cooling system pumps. These changes were noticed in the **Federal Register** on December 27, 2000 (65 FR 81908), August 22, 2001 (66 FR 44170), and August 23, 2001 (66 FR 44382).

Date of issuance: November 2, 2001

Effective date: As of the date of issuance and shall be implemented following refueling outage 17.

Amendment Nos.: 188 and 183

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

Date of initial notice in ***Federal Register:*** December 27, 2000

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 2, 2001.

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: May 30, 2001, as supplemented September 10, 2001.

Brief description of amendments: The amendments change the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.1.3 and adds two new SRs, SR 3.6.1.1.4 and SR 3.6.1.1.5, covering the testing of Suppression Chamber-Drywell Vacuum Breakers and the Drywell-to-Suppression Chamber Bypass Leakage Test.

Date of issuance: November 7, 2001

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

Amendment Nos.: 149 and 135

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

Date of initial notice in ***Federal Register:*** July 25, 2001 (66 FR 38761).

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 amendments are contained in a Safety Evaluation dated November 7, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 9, 2001

Brief description of amendments: These amendments revised the current Technical Specifications of Limerick Generating Station, Units 1 and 2, to make them more consistent with changes to Title 10 of the Code of Federal Regulations, Section 50.59.

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

Effective date: November 1, 2001

Amendment Nos.: 154 and 118

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44170).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 1, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 9, 2001

Brief description of amendments: These amendments replaced the term ''unreviewed safety question'' with ''requires NRC approval pursuant to 10 CFR 50.59'' in order to provide consistency with the changes to 10 CFR 50.59, ''Changes, tests, and experiments,'' which became effective on March 13, 2001.

Date of issuance: November 6, 2001

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

Amendments Nos.: 242 and 246.

Facility Operating License Nos. DPR–44 and DPR–56: The amendments revised the Technical Specifications and the License.

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44170).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 6, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of application for amendment: April 1, 2001, as supplemented July 20, 2001.

Brief description of amendment: This amendment introduces new Technical Specification 6.17, ''Technical Specification (TS) Bases Control Program'' to provide consistency with the changes to 10 CFR 50.59 as published in the **Federal Register** (Volume 64, Number 191) dated October 4, 1999.

Date of issuance: November 15, 2001.

Effective date: As of the date of issuance and shall be implemented within 120 days.

Amendment No.: 249.

Facility Operating License No. NPF–3: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** May 30, 2001 (66 FR 29356).

The supplemental letter contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 15, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 22, 2001.

Brief description of amendments: Revised Technical Specifications Section 6.0, ''Administrative Controls,'' to change the title of the corporate executive responsible for plant nuclear safety from ''President-Nuclear Division'' to ''Chief Nuclear Officer.''

Date of Issuance: November 13, 2001.

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

Amendment Nos.: 178 and 121.

Facility Operating License Nos. DPR–67 and NPF–16: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register***: October 3, 2001 (66 FR 50469).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 13, 2001.

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: June 12, 2000, as supplemented by letters dated November 7, 2000, June 19 and August 17, 2001.

Brief description of amendments: The amendments would use the methodology and the alternative source term (AST) in 10 CFR 50.67 and described in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' and Regulatory Guide 1081, ''Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design-Basis Accidents at Boiling and Pressurized Water Reactors.'' Implementing the AST of 10 CFR 50.67 results in a new acceptance criterion for 10 CFR Part 50, Appendix A, General Design Criterion 19, of 5 rem total effective dose equivalent. The licensee determined that use of the revised analysis assumptions, methodology, and acceptance criterion required prior Nuclear Regulatory Commission (NRC) approval. In addition, the NRC requires in 10 CFR 50.67, a license amendment to implement the AST as a replacement for the Technical Information Document 14844 source term.

Date of issuance: November 13, 2001.

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

Amendment Nos.: 258 and 241.

Facility Operating License Nos. DPR–58 and DPR–74: Amendments approve changes to the updated final safety analysis report.

Date of initial notice in ***Federal Register***: August 23, 2000 (65 FR 51356).

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 amendments is contained in a Safety Evaluation dated November 13, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: February 15, 2001.

Brief description of amendment: The amendment consists of deletion of Operating License Condition 2.D, and revision to the Technical Specifications (TSs) to remove depiction of railroad tracks in TS Figure 4.1–1.

Date of issuance: November 16, 2001.

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

Amendment No.: 190

Facility Operating License No. DPR–46: Amendment revised the Operating License and the Technical Specifications.

Date of initial notice in ***Federal Register***: June 27, 2001 (66 FR 34285).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 16, 2001.

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket Nos. 50–220 and 50–410, Nine Mile Point Nuclear Station, Unit Nos. 1 and 2, Oswego County, New York

Date of application for amendments: February 1, 2001; as supplemented on March 1, March 16, March 29, April 5, April 27, May 30, June 7, September 10, September 26, September 28, and November 2, 2001.

Brief description of amendments: The amendments changed the operating licenses and associated documents to reflect the transfer of Niagara Mohawk Power Corporation's (NMPC's) ownership interest in Nine Mile Point Nuclear Station, Unit No. 1, the transfer of the ownership interests of NMPC, New York State Electric and Gas Corporation, Rochester Gas and Electric Corporation, and Central Hudson Gas & Electric Corporation in Nine Mile Point Nuclear Station, Unit No. 2, and the transfer of NMPC's operating authority for both units, to Nine Mile Point Nuclear Station, LLC. The amendments and corresponding license transfers were approved by the U.S. Nuclear Regulatory Commission by Order dated June 22, 2001, and Supplemental Order dated October 30, 2001.

Date of issuance: November 7, 2001.

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

Amendment No.: 172 (for Unit 1), 100 (for Unit 2).

Facility Operating License Nos. DPR–63 and NPF–69: Amendments revised the operating licenses (both units), Technical Specifications (both units) and Environmental Protection Plan (Unit 2).

Date of initial notice in ***Federal Register***: April 2, 2001 (66 FR 17584).

The staff's related evaluation of the amendments is contained in two Safety Evaluations dated June 22 and October 30, 2001.

No significant hazards consideration comments received: Not applicable.

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: August 16, 2001.

Brief description of amendments: The amendments revised the Technical Specifications by deleting Section 5.5.3, ''Post Accident Sampling,'' and thereby eliminating the requirements to have and maintain the post-accident sampling program. The amendments also revised Section 5.5.2, ''Primary Containment Sources Outside Containment,'' to reflect the elimination of requirements to maintain the post accident sampling system.

Date of issuance: November 13, 2001.

Effective date: As of the date of issuance and shall be implemented on or before June 28, 2002.

Amendment Nos.: 123 and 101.

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

Date of initial notice in ***Federal Register:*** October 3, 2001 (66 FR 50472).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 13, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: August 2, 2001.

Brief description of amendments: The amendments revised the Technical Specifications by deleting Section 6.8.3.d, ''Post Accident Sampling,'' and thereby eliminate the requirements to have and maintain the post-accident sampling program. The amendments also revise Section 6.8.3.a, ''Primary Containment Sources Outside Containment,'' to reflect the elimination of requirements to maintain the post accident sampling system.

Date of issuance: November 7, 2001.

Effective date: As of the date of issuance and shall be implemented within 6 months of the date of issuance.

Amendment Nos.: Unit 1—133; Unit 2—122.

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

*Date of initial notice in **Federal Register:*** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 7, 2001.

No significant hazards consideration comments received: No.

Note: The publication date for this notice will change from every other Wednesday to every other Tuesday, effective January 8, 2002. The notice will contain the same information and will continue to be published biweekly.

Dated at Rockville, Maryland, this 20th day of November 2001.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

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

[FR Doc. 01–29446 Filed 11–27–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Regulatory Guide and Draft Standard Review Plan; Issuance, Availability

The Nuclear Regulatory Commission has issued for public comment a draft of a regulatory guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits and licenses.

The draft guide, temporarily identified as DG–1085 (which should be mentioned in all correspondence concerning this draft guide), is ''Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors.'' DG–1085 is being developed to provide guidance to licensees on the various cost estimates that are required for different stages and methods of decommissioning nuclear power reactors.

A conforming document, Draft NUREG–1713, ''Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors,'' is also being issued for public comment. The NRC staff plans to use Draft NUREG–1713 in their review of licensees' cost estimates for decommissioning that are submitted to the NRC.

The NRC staff is soliciting comments on these draft documents and will incorporate appropriate changes to these documents based on the comments received.

This draft guide and draft standard review plan have not received complete staff approval and do not represent an official NRC staff position.

Comments may be accompanied by relevant information or supporting data. Written comments may be submitted to the Rules and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Copies of comments received may be examined at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by January 30, 2002.

You may also provide comments via the NRC's interactive rulemaking web site through the NRC home page (*http://www.nrc.gov*). This site provides the ability to upload comments as files (any format) if your web browser supports that function. For information about the interactive rulemaking web site, contact Ms. Carol Gallagher, (301) 415–5905; e-mail *CAG@NRC.GOV.* For information about the draft guide and the related standard review plan, contact Mr. W. Mike Ripley at (301) 415–1112; e-mail *WMR@NRC.GOV.*

Although a time limit is given for comments on these drafts, comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time.

Electronic copies of these drafts are available through NRC's interactive rulemaking web site (see above) and from the ADAMS Public Library component on the NRC's web site (the Electronic Reading Room), *http://www.nrc.gov.* These drafts are available for inspection at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415–4737 or (800) 397–4205; fax (301) 415–3548; email *PDR@NRC.GOV.* Requests for single copies of draft or final guides or standard review plans (which may be reproduced), or for placement on an automatic distribution list for single copies of future draft guides in specific divisions, should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section; or by e-mail to *DISTRIBUTION@NRC.GOV;* or by fax to (301) 415–2289. Telephone requests cannot be accommodated. Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them. (5 U.S.C. 552(a))

Dated at Rockville, Maryland, this 8th day of November, 2001.

For the Nuclear Regulatory Commission.

Mabel F. Lee,

Director, Program Management, Policy Development and Analysis Staff, Office of Nuclear Regulatory Research.

[FR Doc. 01–29445 Filed 11–27–01; 8:45 am]

BILLING CODE 7590–01–P

SOCIAL SECURITY ADMINISTRATION

Agreement on Social Security Between the United States and Chile; Entry Into Force

AGENCY: Social Security Administration.

ACTION: Notice.

SUMMARY: The Commissioner of Social Security gives notice that an agreement coordinating the United States (U.S.) and Chilean social security programs will enter into force on December 1, 2001. The agreement with Chile, which was signed on February 16, 2000, is similar to U.S. social security agreements already in force with 18 other countries—Austria, Belgium, Canada, Finland, France, Germany, Greece, Ireland, Italy, Korea (South), Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, and the United Kingdom. Agreements of this type are authorized by section 233 of the Social Security Act.

Like the other agreements, the U.S.-Chilean agreement eliminates dual social security coverage—the situation that exists when a worker from one country works in the other country and is covered under the social security systems of both countries for the same work. When dual coverage occurs, the worker or the worker's employer or both may be required to pay social security contributions to the two countries simultaneously. Under the U.S.-Chilean agreement, a worker who is sent by an employer in one country to work in the other country for 5 years or less remains covered only by the sending country. The agreement includes additional rules that eliminate dual U.S. and Chilean coverage in other work situations.

The agreement also helps eliminate situations where workers suffer a loss of benefit rights because they have divided their careers between the two countries. Under the agreement, workers may qualify for partial U.S. benefits or partial