The meeting on Thursday, September 13, 2001 will be open to the public. If you need special accommodations due to a disability, please contact: Institute of Museum and Library Services, 1100 Pennsylvania Avenue, NW., Washington, DC 20506—(202) 606–8536—TDD (202) 606–8638 at least seven (7) days prior to the meeting date.

Agenda

5th Annual Meeting of The National Museum Services Board and The National Commission on Libraries and Information Science in The Conference Room of Old Sturbridge Village, One Old Sturbridge Village Road, Sturbridge, MA 01566 on Thursday, September 13, 2001

1:30 pm–4:30 pm

I. The Chairs' Welcome and Minutes of the 4th Annual Meeting.
II. Director's Welcome and Opening Remarks.
III. Museum/Library Collaboration: A Case Study.
IV. National Leadership Grants.
  a. Analysis: National Leadership Grants 2001.
  b. Panel and Field Review Process.
  c. Discussion: Emerging Issues and Opportunities.
V. 21st Century Learner: Conference Preview.
VI. National Award for Museum Service/National Award for Library Service.
VII. Budget Update: New Opportunities.

Dated: August 16, 2001.

Linda Bell,

Director of Policy, Planning and Budget, National Foundation on the Arts and Humanities, Institute of Museum and Library Services.

[FR Doc. 01–21313 Filed 8–20–01; 2:11 pm]

BILLING CODE 7036–01–M

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 30, 2001 through August 10, 2001. The last biweekly notice was published on August 8, 2001 (66 FR 41609).

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 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 21, 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 Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). 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

[Docket No. 50–461, Clinton Power Station, Unit 1, DeWitt County, Illinois]

[Docket No. 50–219, Oyster Creek Generating Station, Ocean County, New Jersey]

[Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania]

Date of amendment request: July 9, 2001.

Description of amendment request: The proposed amendments would incorporate TS changes that are being made to provide consistency with the changes to 10 CFR 50.59, ''Changes, tests, and experiments,'' as published in the **Federal Register** (FR) Volume 64, beginning on page 53582 (i.e., 64 FR 53582), dated October 4, 1999. Specifically, the changes replace the terms ''safety evaluation'' with ''10 CFR 50.59 evaluation'' and ''unreviewed safety question'' with ''requires NRC approval pursuant 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. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed changes reflect revision to 10 CFR 50.59, ''Changes, tests, and experiments,'' issued as a Final Rule on October 4, 1999, and do not impact the operation of any system or component assumed in any accident analysis. The proposed changes do not change the requirement to perform a 10 CFR 50.59 review when required by the Technical Specifications Administrative Controls or by a license condition. Due to the administrative nature of these proposed changes there will be no direct impact on the consequences of any accident previously evaluated. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature and do not involve a change to the plant design or operation. No new or different types of equipment will be installed as a result of these changes. The proposed changes make the language in the Technical Specifications Administrative Controls and a license condition conform to the revised 10 CFR 50.59 rule, dated October 4, 1999. No new accident modes or equipment failure modes are created by these proposed changes. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not have a direct effect on any safety analysis assumptions. The proposed changes are administrative in nature and make the Technical Specifications Administrative Controls and a license condition language conform to the revised 10 CFR 50.59 rule, dated October 4, 1999. Changes to the facility that result in meeting the criteria of 10 CFR 50.59 will still require NRC approval pursuant to 10 CFR 50.59. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: December 19, 2000.

Description of amendment request: The proposed amendment would revise the Oyster Creek Technical Specification (TS) Section 3.17 Bases to remove reference to the current licensing basis control room calculated dose consequences and substitute the associated regulatory dose limits that apply for control room habitability in accordance with General Design Criterion 19 and Section 6.4 of the Standard Review Plan. The existing licensing basis control room calculated dose values specified in TS Section 3.17 Bases have been reevaluated as a result of Oyster Creek Licensee Event Report No. 00–006 dated June 26, 2000. This reevaluation has confirmed that the control room habitability dose limits continue to be met. However, this reevaluation is based on use of the U.S. Nuclear Regulatory Commission (NRC)-approved ARCON96 Code methodology for calculation of atmospheric dispersion coefficients (X/Q) for the control room intakes and updated site meteorological data. Incorporation of this new methodology and updated meteorological data into the Oyster Creek licensing basis requires prior NRC review and approval.

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.*

Substitution of the applicable regulatory limits for operator dose in lieu of the specific analyzed values in Technical Specification Section 3.17 Bases is [requested] to be consistent with the existing Technical Specification 4.17 Bases. The proposed change to utilize ARCON96 methodology and updated meteorological data results in control room operator doses that are less than the previously analyzed values, and, therefore, remain within the allowable limits. The probability of accidents is not affected by the computer codes used to assess the consequences of environmental releases. The use of updated, more extensive meteorological data provides a more accurate atmospheric dispersion coefficient (X/Q) value for the Turbine Building release to the control room ventilation system air intake.

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

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

The change to incorporate ARCON96 methodology and updated meteorological data for assessing the control room operator doses from the releases of radioactive material following an accident has no [e]ffect on creating a new or different kind of accident. The proposed change does not affect the operation or functionality of any structures, systems or components.

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 change does not involve a significant reduction in a margin of safety.*

The proposed change involves [a] revision to Technical Specification Section 3.17 Bases to substitute applicable regulatory limits in lieu of the specific analyzed dose values. The proposed change to incorporate ARCON96 methodology and updated meteorological data results in a more accurate determination of conservative control room air intake X/Q values and the resulting control room operator dose. ARCON96 is an NRC approved methodology which provides an acceptable level of conservatism. The updated meteorological data [are] obtained in accordance with NRC Regulatory Guide 1.23 requirements.

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: Richard P. Correia, Acting.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: May 29, 2001.

Description of amendment request: The proposed amendment would revise the Haddam Neck Plant Defueled Physical Security Plan referenced in License Condition 2.C(5). The proposed amendment reflects the intent of Connecticut Yankee Atomic Power Company (CYAPCO) to transfer all spent nuclear fuel and Greater than Class C waste from wet storage in the spent fuel pool to dry casks located at an on-site Independent Spent Fuel Storage Installation (ISFSI). CYAPCO proposed to make this ISFSI Security Plan an attachment to the existing Defueled Physical Security Plan. Adding the ISFSI Security Plan as an attachment to the Defueled Physical Security Plan would enable CYAPCO to implement the ISFSI Security Plan portion prior to commencement of fuel transfer operations.

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

The proposed amendment to the Security Plan provides the basis for establishing security functions necessary to implement appropriate security/safeguards measures for the CYAPCO Independent Spent Fuel Storage Installation (ISFSI). As such, the changes will not:

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

The proposed amendment to the Security Plan, which incorporates ISFSI security functions, does not reduce the ability of the Security organization to prevent radiological sabotage and, therefore, does not increase the probability or consequences of a radiological release previously evaluated. The proposed Security Plan changes will not affect any important to safety systems or components, their mode of operation or operating strategies. The proposed Security Plan changes have no affect on accident initiators or mitigation. Therefore, the proposed amendment to the Security Plan will 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 proposed amendment of the Security Plan incorporating ISFSI security functions does not affect the operation of systems important to safety. The Security Plan amendment does not affect any of the parameters or conditions that could contribute to the initiation of any accident. No new accident scenarios are created as a result of Security Plan changes requested to incorporate the ISFSI security functions. In addition, the design functions of equipment important to safety are not altered as a result of the proposed Security Plan changes. Therefore, the proposed Security Plan changes will not create the possibility of a new or different accident from any previously evaluated.

3. *Involve a significant reduction in the margin of safety.* Implementation of the proposed amendment to the Security Plan incorporating ISFSI security functions will not reduce a margin of safety as detailed in the Technical Specifications as there are no Technical Specification requirements associated with the physical security system. Specifically, the proposed changes to the Security Plan do not represent a change in initial conditions, system response time, or in any other parameter affecting the course of an accident analysis supporting the Basis of any Technical Specification. The proposed amendment to the Security Plan does not reduce the effectiveness of any security/safeguards measures currently in place at CYAPCO. Therefore, the proposed Security Plan changes will not involve a significant reduction in the margin of safety.

Based on the considerations noted above, it is concluded that the proposed changes will not endanger the public health and safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the 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. Robert K. Gad, III, Ropes & Gray, One International Plaza, Boston, Massachusetts 02110–2624.

NRC Section Chief: Stephen Dembek.

Connecticut Yankee Atomic Power Company, Docket No. 50–213, Haddam Neck Plant, Middlesex County, Connecticut

Date of amendment request: May 30, 2001.

Description of amendment request: The proposed amendment would correct terminology, clarify the specification for consistency with established programs and Standard Technical Specifications, (TSs) and reflect current plant conditions. The proposed changes also reflect the current organization titles. The licensee also proposed changes to the TS Bases for spent fuel pool water level and cooling.

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

CYAPCO has reviewed the proposed changes to the Operating License and the Technical Specifications in accordance with 10 CFR 50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. An evaluation against these standards is provided below as first a summary against the overall change, and also against each of the specific proposed changes.

The proposed changes do not involve an SHC because the changes would not:

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

In the present plant configuration, the reactor-related accidents previously evaluated (i.e., LOCA, MSLB, etc.) are no longer possible. The accidents previously evaluated that are still applicable to the plant are fuel handling accidents and gaseous and liquid radioactive releases. The following events are presently considered as bounding of all other events:

- Fuel handling and cask drop accidents in the spent fuel building,
- Criticality in the spent fuel pool,
- Loss of spent fuel cooling,
- Resin fire (gaseous release), and
- Rupture of a tank containing radioactive liquid.

There is no significant increase in the probability of a fuel handling accident since refueling operations have ceased, with a corresponding decrease in the frequency of fuel movement. The radiological consequences of a fuel handling accident, should one occur, decrease the longer the spent fuel is allowed to decay. The spent fuel inventory of radioactive iodine and noble gases have decayed more than 20 half-lives since shutdown and are no longer a release concern. The allowed weight over the spent fuel pool is still less than that previously approved. Therefore, there has been no increase in the probability or consequences of a fuel handling or cask drop accident.

Criticality controls are imposed by specifications 3/4.9.13 and 3/4.9.14. There have been no technical changes to these specifications. Therefore, there has been no increase in the probability or consequences of a criticality event.

Spent fuel cooling is maintained by keeping the pool temperature below 150°F. Should normal cooling be lost, the availability of an abundant supply of water ensures that sufficient time is available to restore cooling. This is controlled by specifications 3/4.9.11 and 3/4.9.16. There have been no technical changes to these specifications. Therefore, there has been no increase in the probability or consequences of a loss of cooling event.

The probability of a gaseous or liquid radioactive release is not changed by the proposed revisions. As the plant undergoes decommissioning, the previous limiting events are no longer applicable, and previous non-limiting events now become limiting. These new events have not changed from how they might have occurred in the past. The radiological consequences of a gaseous or liquid radioactive release are bounded by the fuel handling accident during defueled operation and a spent resin fire during processing of resin from the reactor coolant system decontamination. The rupture of a tank containing radioactive liquid was assessed and found to be bounded by these events. With the plant defueled and permanently shutdown, the demands on the radwaste systems are lessened since no new radioisotopes are being generated by irradiation or fission. Therefore, there is no increase in the probability or consequences of a gaseous or liquid radioactive release.

The changes to conform to Section 6.0 to draft NUREG–1625 are of an administrative nature, and have been reviewed and found to be safe.

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

The proposed changes are generally of an administrative nature and do not have an effect on the physical plant. The events considered bound other potential events and are considered the limiting cases for potential gaseous or liquid releases to the environment.

With the plant undergoing decommissioning, the types of accidents one might be concerned with involve criticality of the spent fuel, or draining of the spent fuel pool. None of the proposed changes affect the possibility of such an event. Also, none of the proposed changes could lead to a radiological release of a greater magnitude than for the events considered, such as might occur with the accumulation of a greater quantity of radioactive material in one location, or with damage to a greater number of fuel assemblies than considered in the fuel handling accident.

The proposed changes do not affect systems, structures and components and have no adverse impact on the storage of fuel nor on the processing of radioactive wastes presently at the site. The present set of limiting events is a subset of events previously considered. Therefore these changes do not create the possibility of a new or different kind of accident from any accident previously considered.

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

The proposed changes do not reduce a margin of safety because there is no direct affect on any safety analysis assumptions. Changes to the Technical Specifications Bases reflect current plant conditions.

Based on the above evaluation, CYAPCO concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

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

Attorney for licensee: Mr. Robert K. Gad, III, Ropes & Gray, One International Plaza, Boston, Massachusetts 02110–2624.

NRC Section Chief: Stephen Dembek.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: July 13, 2001.

Description of amendment request: The proposed amendment would make a one-time change to Technical Specification Surveillance Requirement 4.4.A.3 to revise the frequency for the containment integrate leak rate test (ILRT, Type A test) from at least once per 10 years to once per 15 years. The change would apply only to the interval following the last Type A test that was satisfactorily performed in June 1991 at Indian Point Nuclear Generating Unit No. 2 (IP2).

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 of occurrence or consequences of an accident previously evaluated.*

The change does not affect the ability of the containment to mitigate the consequences of an accident. The containment is not an accident initiating system or structure. The proposed one time change to Type A testing frequency has been determined to be adequate as documented in NUREG–1493 [''Performance-Based Containment Leak-Test Program,'' September 1995] which determined generically that very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. This generic result has been confirmed for IP2 by a plant specific risk impact assessment. Past IP2 Type A tests show leakage to be below acceptance criteria, indicating a very leak-tight containment, without credit for the weld channel and penetration pressurization system (WC&PPS). Inspections required by other TS and by the ASME [American Society of Mechanical Engineers] code are performed in order to identify indications of containment degradation that could affect that leak tightness. The WC&PPS monitors the leak tightness of liner plate welds in the containment during plant operation as required by Technical Specifications. Type B and C testing required by TS will identify any containment opening such as valves that would otherwise be detected by the Type A tests. The frequency of performance of surveillance does not result in any hardware changes or the response of equipment in performing its specified function. Therefore, 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.

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 does not introduce nor increase the number of failure mechanisms of a new or different type of accident than those previously evaluated since there are no physical changes being made to the facility. Performance of the testing on the revised schedule will not have an adverse affect on the ability of the containment to perform its intended function. The proposed change does not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. Therefore, the 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 the margin of safety.*

The one time change to the current frequency for Type A testing still provides adequate assurance of containment integrity. The NUREG–1493 generic study of the effects of extending containment leakage testing found that a 20-year extension in Type A leakage testing resulted in an imperceptible increase in risk to the public. NUREG -1493 found that, generically, the design containment leakage rate contributes about 0.1 percent to the individual risk and that the decrease in Type A testing frequency would have a minimal affect on this risk since 95% of the potential leakage paths are detected by Type B & C testing. The risk impact change of the test frequency was small. Online testing of the integrity of liner plate welds using the WC&PPS and regular inspections will further reduce the risk of a containment leakage path going undetected. There are no changes being made to TS safety limits or safety system settings that would adversely affect plant safety. Therefore, operation of the facility in accordance with the proposed amendment would 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: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Richard P. Correia, Acting.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: July 13, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 6.12, ''High Radiation Area,'' to delete the administrative requirements for the control of access to high radiation areas. The control of access to these areas is assured by the licensee's radiation protection programs that comply with 10 CFR 20.1601 by using the alternate methods in NRC Regulatory Guide 8.38, ''Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,'' June 1993.

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 in the consequences of an accident previously evaluated?*

The proposed TS change is administrative in nature. It involves deleting specific requirements for complying with a subparagraph of 10CFR20 for the purpose of controlling access to high radiation areas. Accident evaluations do not consider the effects of methods of controlling access to high radiation areas. The proposed changes do not result in a change to the design or operation of [...] any plant structure, system, or component. Therefore any assumptions of the operability or performance of any structure, system, or component in accident evaluations are unchanged.

Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed change is administrative in nature. The methods of controlling access to high radiation areas do not affect the design or operation of any plant structure, system, or component. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed TS change is administrative in nature. It involves deleting specific requirements for complying with a subparagraph of 10CFR20. However, effective compliance with 10CFR20 is mandated by the IP2 [Indian Point 2] Facility Operating License Section C. The effectiveness of Con Edison compliance with 10CFR20 is not adversely affected by this change. In addition, this change does not affect any design function for or the operation of any plant structure, system, or component.

Therefore, the change does not affect any of the safety analyses or any 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: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Richard P. Correia, Acting.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of amendment request: July 13, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to delete TS Tables 3.6–1, ''Non-Automatic Containment Isolation Valves Open Continuously or Intermittently for Plant Operation,'' and 4.4–1, ''Containment Isolation Valves.'' The proposed amendment would also revise other TS sections that reference these tables. The removal of the tables is in accordance with the guidance in NRC Generic Letter (GL) 91–08, ''Removal of Component Lists from Technical Specifications.''

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 in the consequences of an accident previously evaluated?*

The proposed changes consist of removal of the containment isolation valve component lists from the IP2 [Indian Point 2] TS and corresponding editorial changes to support removal of the tables. The changes are being made in accordance with the guidance provided by the NRC in GL 91–08 and do not alter existing TS requirements or those components to which the TS requirements apply. The information contained in the Tables being removed is duplicated in the UFSAR [Updated Final Safety Analysis Report] and other appropriate plant procedures. Any subsequent changes regarding the individual components or their operation would be evaluated under the requirements of 10CFR50.59. The proposed changes do not involve a change to the design or operation of any plant structure, system, or component. Nor are the safety analyses affected as a result of the changes. Accordingly, the initiators of any accident as well as any structure, system or component relied upon for the mitigation of the accident are not affected by the proposed changes.

Therefore, there is no increase in the probability or in the consequences of an accident previously evaluated.

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

The proposed changes do not involve a change to the design or operation of [...] any plant structure, system or component. The proposed changes involve the removal of component lists for containment isolation valves from the TS. In accordance with the guidance provided by GL 91–08, the conditions, actions, and requirements of the TS will apply to those valves that are classified as containment isolation valves by the plant licensing basis. This includes the testing of Containment Isolation Valves as required by 10CFR50 Appendix J and IP2 TS 4.4.D.1.a. Required specifications and requirements of the tables remain applicable. There are no changes to any parameter used in the accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident for any previously evaluated.

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

The proposed changes are in accordance with the guidance provided by the NRC in GL 91–08 and NUREG–1431, Standard Technical Specifications. The changes will maintain current safety margins while reducing the regulatory and administrative burdens to both the NRC and IP2. The proposed changes will not result in changes to the design or operation of any plant system and do not involve changes to any 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: Brent L. Brandenburg, Esq., 4 Irving Place, New York, New York 10003.

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: July 31, 2001.

Description of amendment request: The proposed amendment requests U.S. Nuclear Regulatory Commission (NRC) approval of Big Rock Point Plant's (Big Rock Point) Security Plan, Suitability Training and Qualification Plan, and Safeguards Contingency Plan. These plans reflect the addition of provisions relating to the loading and storage of spent nuclear fuel at the Independent Spent Fuel Storage Installation (ISFSI).

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

The proposed change does not:

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

The currently approved and implemented Security Plan (Defueled Security Plan) is not being changed. The ISFSI Security Plan is being added to the scope of the overall security plan for the Big Rock Point site. The additions to the overall [Security] Plan have been evaluated in accordance with 10 CFR 50.54(p) and 10 CFR 72.212(b)(4) and it has been determined that the implementation of the ISFSI Security Plan would not decrease the effectiveness of the Defueled Security Plan, the Defueled Suitability Training and Qualification Plan, or the first four categories of the Defueled Safeguards Contingency Plan.

The ISFSI Security Program staffing will be parallel to the staffing requirements of the Defueled Security Plan, except that one Central Alarm Station [CAS] operator will be employed during the period when spent fuel is located in the spent fuel pool in the plant and also located in dry fuel storage at the ISFSI facility.

The operational and physical venues of the Defueled Security Plan and the ISFSI Security Plan are separate and distinct, except for the utilization of a single CAS operator, and the lines of demarcation between the two plans [are] clearly defined and not overlapping. The implementation of any of the plans does not therefore degrade or inhibit the implementation of the other plan.

The Defueled Suitability Training and Qualification Plan and the Defueled Safeguards Contingency Plan also have not been changed. A separate and parallel ISFSI Training and Qualification Plan and ISFSI Contingency Plan is included in the ISFSI Security Plan. The physical protection systems described in the ISFSI Plans are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106.

Therefore, the ISFSI Plan revisions of the Big Rock Point Plant Security Plan, Suitability Training and Qualification Plan and the Safeguards Contingency Plan will not increase the probability or the consequences of an accident previously evaluated since the previously approved Defueled Suitability Training and Qualification Plan and the Safeguards Contingency Plan remain unchanged.

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

The ISFSI Security Plan has no impact on the existing Defueled Security Plan since they operate in different physical and licensing venues. The accidents considered for the Spent Fuel Pool, the venue of the Defueled Security Plan, are described in the Big Rock Point Updated Final Hazards Summary Report. The accidents considered for the ISFSI are contained in the FuelSolutions Final Safety Analysis Reports [FSARs] for the W150 Storage Cask, W100 Transfer Cask and the W74 Canister under Docket No. 72–1026.

The ISFSI Security Plan has been crafted to meet or exceed all of the assumptions of the FuelSolutions FSARs concerning accident analyses and the plan meets or exceeds all of the applicable requirements of 10 CFR 73.55 with approved exceptions or approved alternative measures. The physical protection systems described in the ISFSI Security Plan are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106.

The proposed action does not affect plant systems, structures or components within the venue of the existing Defueled Security Plan. The ISFSI additions to the Security Plan, Suitability Training and Qualification Plan and the Safeguards Contingency Plan do not create the possibility of a new or different kind of accident from any accident previously evaluated since the previously approved Defueled Security Plan, [Suitability] Training and Qualification Plan and Safeguards Contingency Plan remain the same.

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

The addition of a separate, parallel ISFSI Security Plan, Suitability Training and Qualification Plan, and Safeguards Contingency Plan does not alter or reduce the effectiveness of the previously approved Defueled Security Plan. The physical protection systems described in the ISFSI Plan are designed to protect against the loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose as described in 10 CFR 72.106. Therefore, the margin of safety will not be reduced as a result of the ISFSI addition to the Security Plan, or an ISFSI specific addition of a Suitability Training and Qualification Plan or an ISFSI specific addition of a Safeguards Contingency Plan.

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

Attorney for licensee: David A. Mikelonis, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 31, 2001.

Description of amendment request: The proposed amendment would revise and relocate the inservice testing portion of Technical Specification (TS) 5.0.5 to TS 6.5.8, and eliminate the inservice inspection portion of TS 4.0.5. In addition, other sections of the TSs that reference TS 4.0.5 would be revised to be consistent with the revisions discussed above.

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 operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequence of an accident previously evaluated?*

The proposed change relocates the requirements to test and inspect ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code] Class 1, 2, and 3 components from TS 4.0.5 to the administrative section of the TSs and includes modifications to the wording to make it consistent with NUREG–1432 [Standard Technical Specifications, Combustion Engineering Plants]. This change will not reduce the current testing and inspection requirements. The performance of a code inservice test is not an accident initiator. The proposed change for removing the statement for NRC [Nuclear Regulatory Commission] granting written relief for [from the] ASME Code does not involve a significant increase in the probability or consequences of an accident. Verbally issuing relief to the ASME Code by the NRC does not reduce assurance of the health and safety of the public since the NRC still reviews the basis for the relief on its technical merit and the NRC Staff still obtains management approval prior to granting the relief.

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

2. *Will 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?*

[The citation at] 10 CFR 50.55a, "Codes and Standards" governs inservice testing and inspection requirements. The inspection requirements contained in 10 CFR 50.55a paragraph (g) are duplicated in TS 4.0.5. This duplication is unnecessary and therefore, the wording related to the inspection requirements will be deleted in the proposed change. No actual change to the inspection or testing activities are proposed as the requirements in 10 CFR 50.55a continue to govern these. Therefore, the testing and inspection requirements will remain the same as those presently required. The proposed change is administrative in nature in that it relocates testing requirements from one section of the TSs to another and modifies the wording to be consistent with NUREG–1432. The removal of the requirement to obtain written relief from the NRCc staff will not create the possibility of any new or different types of accidents. Staff review is still required prior to granting the relief.

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

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

The testing and inspection requirements contained in TS 4.0.5 are governed by 10 CFR 50.55a, "Codes and Standards." The 10 CFR requirements to perform the ASME code testing and inspections will not be reduced by the proposed change. The inspection and tests will continue to be performed as they are currently. This change moves the present requirements from one section of the TSs to another.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 31, 2001.

Description of amendment request: The proposed amendment would revise the technical specifications (TSs) to allow an extension of the three-year inspection interval of the reactor coolant pump flywheel voumetric examination to ten years. In addition, the requirement discussed above would be moved to the administrative controls section of the TSs.

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

1. *Will operation of the facility in accordance with this proposed change involve a significant increase in the*

probability or consequences of an accident previously evaluated?

Inspections of the reactor coolant pump (RCP) flywheels are conducted to detect a flaw in the flywheel prior to it becoming a missile that could damage other portions of the facility. The fracture mechanics analyses conducted as part fo the NRC [U.S. Nuclear Regulatory Commission] approved Topical Report SIR–94–080–A, Rev. 1, shows that a conservatively sized pre-existing crack will not grow to a flaw size necessary to create flywheel missiles with the current or extended life of the facility. This analysis conservatively assumes minimum material properties, maximum flywheel speed, location of the flaw in the highest stress area, and a number of startup and shutdown cycles higher than expected. Since a conservative flaw in the RCP flywheels will not grow to the allowable flaw size under large break LOCA [loss-of-coolant accident] conditions over the life of the plant, reducing the inspection frequency of the flywheels will not significantly increase the probability or consequences of an accident previously evaluated.

The change to move the survillance requirements for the RCP flywheels to the programs section of the technical specifications is administrative and has no impact on probability or consequences of an accident.

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

2. *Will 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?*

The proposed changes will not alter the plant configuration or require any new or usual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes.

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

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

The ANO–2 [Arkansas Nuclear One, Unit 2] flywheels are made of either ASTM [American Society for Testing and Materials] A–533, Grade B, Class 1 or A–508, Class 5 steel plate material, which is pressure vessel quality steel. These materials have high tensile and yield strength qualities. The operating temperature of the flywheel is not less than 100 °F and the RT_{NDT} value is below +10 °F. Therefore, there is at least 90 °F margin below the lowest temperature at which operating speed is achieved which is in accordance with Regulatory Guide 1.14, Rev. 1, ''Reactor Coolant Pump Flywheel Integrity.'' The fracture mechanics analyses conducted to support the extension of the inspection frequency from 3 to 10 years was performed with substantial conservatism built into the analyses. Even with this analytical conservatism, the results indicate that the flywheels have sufficient margin that there is only a negligible potential for gross failure of the flywheels.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: July 10, 2001.

Description of amendment request: Technical Specification (TS) Surveillance Requirement (SR) 4.8.1.1.2.e requires certain emergency diesel generator (EDG) surveillances be performed during shutdown. The proposed change will modify this SR to allow performance of specific surveillances during any mode of plant operation. This will provide flexibility in the scheduling of testing activities consistent with online maintenance activities and improve EDG availability during plant shutdown periods.

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

The EDG is designed to operate in the event of a loss of offsite power or upon receipt of a SIAS [Safety Injection Actuation Signal]. No modifications or design changes are proposed to the EDG in conjunction with this proposed TS change. Periodic testing of the EDG starting circuitry, lockout relays, capability to reject a load and maintain voltage and frequency, ability to run for 24-hours, and various other tests prove the EDG is qualified to function upon demand. The changes proposed will allow several SRs to be performed in modes other than only during shutdown. A review of each of these has been performed. The system alignment needed to achieve these tests is the same whether the test is performed during shutdown or during power operations. When performing SR 4.8.1.1.2.e.1, 2, 4, 6, and 9, the EDG is operable and capable of performing its intended function, if called upon. When performing SR 4.8.1.1.2.e.10 and 12, the EDG that is being tested is inoperable for less than two hours, which is well within the allowable outage time. While performing these SRs, operations personnel are available to quickly respond to align the EDG as needed for an unexpected event. Additionally, the equipment covered by these specifications are not accident initiators and can not cause an accident.

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

2. *Will 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?*

The proposed change does not impact any system or component which could cause an accident. The proposed change will not alter the plant configuration (no system design modifications are required) or require any unusual operator actions. The proposed change will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. A review of the proposed change indicates that the required testing will be performed in a similar configuration and the interrelationship with other components is the same whether the testing is performed at power or during shutdown. The proposed change does not introduce any new failure modes. Additionally, the response of the plant and the operators following an accident will not be significantly different as a result of these changes.

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

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

The proposed TS change is associated with the surveillance requirements for the EDGs. The proposed change allows certain EDG surveillance requirements to be performed when the plant is at power rather than when shutdown. When performing SR 4.8.1.1.2.e.1, 2, 4, 6, and 9, the EDG is operable and capable of performing its intended function, if called upon. When performing SR 4.8.1.1.2.e.10 and 12, the EDG that is being tested is inoperable for less than two hours, which is well within the allowable outage time. The proposed change will have no adverse effect on plant operation or equipment important to safety. The plant response to the design basis accidents will not change and the accident mitigation equipment will continue to function as assumed in the design basis accident analysis.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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.

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

Date of amendment request: July 23, 2001.

Description of amendment request: This submittal requests a change to administrative Technical Specification (TS) 6.15. The change postpones the next Type A test performed after May 12, 1991, to no later than May 11, 2006, which basically results in an extended interval of 15 years for performance of the next Integrated Leak Rate Test (ILRT).

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

[Appendix J of 10 CFR [Part] 50], was amended to incorporate provisions for performance-based testing in 1995. The proposed amendment to Technical Specification (TS) 6.15 adds a one-time extension to the current interval for Type A testing (*i.e.,* the integrated leak rate test). The current interval of ten years, based on past performance, would be extended on a one-time basis to 15-years from the date of the last test. The proposed extension to the Type A test cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since research documented in NUREG–1493 has found that, generically, fewer than 3% of the potential containment leak paths are not identified by Type B and C testing. Waterford 3 [Waterford Steam Electric Station, Unit 3], through testing and containment inspections, also provides a high degree of assurance that the containment will not degrade in a manner detectable only by a Type A test. Inspections required by the Maintenance Rule (10 CFR 50.65) and by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leaktightness.

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

2. *Will 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?*

The proposed extension to the interval for the Type A test does not involve any design or operational changes that could lead to a new or different kind of accident from any accidents previously evaluated. The test itself is not changing and is just to be performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

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

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

The generic study of the increase in the Type A test interval, NUREG–1493, concluded there is an imperceptible increase in the plant risk associated with extending the test interval out to twenty years. Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detect 97% of potential leakage paths. For the requested change in the Waterford 3 ILRT interval, it was determined that the risk contribution of leakage will increase 0.17%. This change is considered very small and does not represent a significant reduction in the margin of safety.

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

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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.

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

Date of amendment request: July 23, 2001.

Description of amendment request: The proposed change is to delete Technical Specifications (TS) 3.9.12, ''Fuel Handling Building Ventilation System,'' and TS 3.3.3.1 requirements for the Fuel Storage Pool area radiation monitors.

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

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

The FHBVS [Fuel Handling Building Ventilation System] is not involved in the initiation of any accidents. The system is not credited with providing any supplemental filtration of any releases from an accident occurring in the containment building. It was designed to provide an accident mitigation function by isolating the system and filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident (FHA). The charcoal adsorber was the primary component that supported this filtration function. However, based on a revised analysis of the dose consequences of the FHA, it has been demonstrated that doses due to the FHA, to both the public and the control room operator, remain well within regulatory acceptance limits even assuming no credit for either isolation or filtration. The charcoal filtration function is not required and need not be tested. Thus, there is no required safety function in the event of a fuel handling accident provided by either the ventilation system or the area radiation monitor.

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

2. *Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.*

The FHBVS is not involved in the initiation of any accidents. It was designed to provide an accident mitigation function by isolating the system and filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident (FHA). Recent analyses show that the isolation and filtration functions are no longer required. The charcoal adsorber can not influence any accident initiators. Further, it has been demonstrated that the deletion of the technical specification requirements does not impact this conclusion and does not influence any new potential accident scenarios in any way.

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

3. *Does Not Involve a Significant Reduction in the Margin of Safety.*

The FHBVS was designed to provide an accident mitigation function by filtering the radioiodines that may be released from a damaged fuel assembly in the event of a Fuel Handling Accident (FHA). Charcoal adsorbers had been provided for this function. Recent analysis of the FHA in the Fuel Handling Building demonstrate that the isolation function and the charcoal adsorber are not required to satisfy the margin of safety provided by the Technical Specification requirements. Based on a revision to the dose consequence analysis of the FHA, it has been determined that doses remain well within the regulatory allowable for exposure even assuming no credit for charcoal filtration. The margin of safety, as defined by SRP [Standard Review Plan] 15.7.4, Rev 1, and General Design Criterion 19, has not been significantly reduced.

Therefore, the proposed changes do 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: 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]

[Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania]

[Docket Nos. STN 50–277 and STN 50–278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania]

[Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station, Units 1 and 2, Lake County, Illinois]

Date of amendment request: July 9, 2001.

Description of amendment request: The proposed amendments would incorporate Technical Specifications (TS) changes that are being made to provide consistency with the changes to 10 CFR 50.59, ''Changes, tests, and experiments,'' as published in the **Federal Register** (FR) Volume 64, beginning on page 53582 (i.e., 64 FR 53582), dated October 4, 1999. Specifically, the changes replace the terms ''safety evaluation'' with ''10 CFR 50.59 evaluation'' and ''unreviewed safety question'' with ''requires NRC approval pursuant to 10 CFR 50.59.''

In addition, Exelon proposes to change a condition 3.B of Operating License Nos. DPR–44 and DPR–56 for the Peach Bottom Atomic Power Station, Units 2 and 3.

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

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

The proposed changes reflect revision to 10 CFR 50.59, ''Changes, tests, and experiments,'' issued as a Final Rule on October 4, 1999, and do not impact the operation of any system or component assumed in any accident analysis. The proposed changes do not change the requirement to perform a 10 CFR 50.59 review when required by the Technical Specifications Administrative Controls or by a license condition. Due to the administrative nature of these proposed changes there will be no direct impact on the consequences of any accident previously evaluated. Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes are administrative in nature and do not involve a change to the plant design or operation. No new or different types of equipment will be installed as a result of these changes. The proposed changes make the language in the Technical Specifications Administrative Controls and a license condition conform to the revised 10 CFR 50.59 rule, dated October 4, 1999. No new accident modes or equipment failure modes are created by these proposed changes. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not have a direct effect on any safety analysis assumptions. The proposed changes are administrative in nature and make the Technical Specifications Administrative Controls and a license condition language conform to the revised 10 CFR 50.59 rule, dated October 4, 1999.

Changes to the facility that result in meeting the criteria of 10 CFR 50.59 will still require NRC approval pursuant to 10 CFR 50.59. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the 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 Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Date of amendment request: September 29, 2000, as supplemented by letter dated March 1, 2001 (previously noticed in the **Federal Register** on December 27, 2000, 65 FR 81908).

Description of amendment request: The March 1, 2001, supplement requests an amendment to revise the technical specifications to (1) increase the number of required automatic depressurization system (ADS) valves from four to five, (2) add surveillance requirements for the operability of the additional ADS valve, (3) change a surveillance requirement to verify the flow rate of two low-pressure coolant injection pumps instead of three pumps, consistent with the accident analyses, and (4) remove an allowance to continue operating for 72 hours if certain combinations of emergency core cooling system systems are inoperable. These are additional changes to those that were requested in the September 29, 2000, application. The changes to the technical specifications support a change in fuel vendors from Siemens Power Corporation to General Electric (GE) and a transition to the use of GE–14 fuel.

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

The proposed changes do not affect the initiators of analyzed events or the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems or components. The proposed changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any events. The evaluations supporting the transition to GE fuel revealed that the current Technical Specification (TS) Limiting Condition for Operation (LCO) and conditions must be revised to place additional limitations on equipment to ensure that the plant is operated within the assumptions of the safety analyses. With the additional limitations, the analyses demonstrate that all of the acceptance criteria continue to be met. As a result, the changes do not involve a significant increase in the probability of consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the facility or change the normal facility operation. No new or different equipment is being installed and no installed equipment is being removed. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the changes therefore do not increase the possibility of a new or different kind of accident from any previously evaluated.

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

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 setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. The changes reflect a reduction in redundancy in the capability of the Automatic Depressurization System (ADS)[.] However, the proposed changes impose more restrictive requirements on operation to ensure that all of the accident analyses continue to meet acceptance criteria. Therefore the proposed changes do not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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 Energy Company, LLC, Docket Nos. 50–352 and 50–353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: May 23, 2001.

Description of amendment request: Exelon proposed changes that would delete Action Statement b. associated with Limiting Condition for Operation 3.4.2 regarding operations with a stuck open safety/relief valve.

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

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

The proposed TS change deletes Action Statement b. associated with Limiting Condition for Operation (LCO) 3.4.2 concerning plant operations with stuck open safety/relief valves. The operator action described in the LCO represents detailed methods of responding to an event, and therefore, if eliminated, would not result in increasing the probability of the event nor act as an additional initiator of an event. Therefore, this action can be eliminated, and will not involve a significant increase in the probability of an accident previously evaluated.

As discussed in Section 15.1.4 (''Inadvertent Main Steam Relief Valve Opening''), of the Limerick Generating Station, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR), a main steam relief valve is postulated to inadvertently open. While this transient does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via relief valve operation. Because this activity is contained within the primary containment, there is no exposure to operating personnel or uncontrolled release of radioactivity to the environment. Therefore, this change does not increase the consequences of an accident previously evaluated.

The requirement to scram the reactor within 2 minutes of identifying a stuck open safety/relief valve was not incorporated into the BWR Standard Technical Specifications (NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 1, dated April 1995).

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 TS change deletes Action Statement b. associated with Limiting Condition for Operation 3.4.2 concerning safety/relief valves. This change does not change the design or configuration of the plant. The safety/relief valves are accident mitigators. Section 15.1.4 (''Inadvertent Main Steam Relief Valve Opening''), of the Limerick Generating Station, Units 1 and 2, Updated Final Safety Analysis Report (UFSAR), postulates an inadvertent opening of a main steam relief valve. This change will not alter the assumptions or results of this analysis. No new operation or failure modes are created, nor is a system-level failure mode created that is different than those that already exist. Therefore, this 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 does not involve a significant reduction in a margin of safety, nor does it affect any analytical limits. There are no changes to accident or transient core thermal hydraulic conditions, or fuel or reactor coolant boundary design limits, as a result of the proposed change. The proposed change will not alter the assumptions or results of the analysis contained in the UFSAR. 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: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 26, 2001.

Description of amendment request: The proposed amendment would revise the Limerick Generating Station (LGS) Units 1 and 2, Technical Specifications (TSs) 3/4.3.3, Actions 36 and 37 of Table 3.3.3–1, and the associated TS Bases. The change to Action 36 clarifies equipment affected by inoperable components. The change to Action 37 takes advantage of the inherent overlap of the degraded voltage relays' characteristics such that inoperable relays that define a channel can be taken out of service without placing its associated source breaker in the trip position.

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

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

The changes to Action 36 detail what equipment is impacted by an inoperable bus under voltage relay. Making these changes assures that the appropriate equipment is considered inoperable. Identifying the impacted equipment for an inoperable under voltage relay does not increase the probability or the consequences of an accident previously evaluated.

Action 37 presently requires placing an inoperable channel (relay) in the tripped condition which results in making the associated offsite source circuit breaker unavailable to that bus. Changing Action 37 to place a relay in the bypass condition, rather than the tripped condition, permits the offsite source of power to still be available to the bus in the event of an inoperable degraded voltage relay. The change to Action 37 takes advantage of the inherent overlap of the degraded voltage relays' characteristics such that inoperable relays that define the channel can be taken out of service without placing its associated source breaker in the trip position. The change to Action 37 does not adversely impact the availability or reliability of the offsite power system. Therefore, the proposed changes to Action 37 do not increase the probability or consequences of an accident previously evaluated.

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

The changes to Action 36 detail what equipment is impacted by an inoperable bus under voltage relay and does not involve physical changes to the plant that would create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to Action 37 take advantage of the overlap of the degraded voltage relays by providing actions to be taken when an individual relay within a channel is inoperable. Changing Action 37 does not make any physical changes to the plant. Therefore, the changes to Action 37 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The changes to Action 36 and Action 37 do not affect the availability or operation of mitigation systems. Therefore, there is no impact on event analysis that would affect the resultant analyses or reduce 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.

Exelon Generation Company, LLC, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: September 29, 2000, as supplemented by letter dated March 1, 2001 (previously noticed in the **Federal Register** on December 27, 2000, 65 FR 81912).

Description of amendment request: The March 1, 2001, supplement requests an amendment to revise the technical specifications to increase the number of required automatic depressurization system (ADS) valves from four to five, to add surveillance requirements for the operability of the additional ADS valve, and to remove an allowance to continue operating for 72 hours if certain combinations of emergency core cooling systems are inoperable. These are additional changes to those that were requested in the September 29, 2000, application. The changes to the technical specifications support a change in fuel vendors from Siemens Power Corporation to General Electric (GE) and a transition to the use of GE–14 fuel.

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

The proposed changes involve more restrictive limitations on operation. These changes do not affect the initiators of analyzed events or the assumed mitigation of accident or transient events. Analyzed events are initiated by the failure of plant structures, systems or components. The proposed changes do not impact the condition or performance of these structures, systems or components. Consequences of analyzed events are the result of the plant being operated within assumed parameters at the onset of any events. The evaluations supporting the transition to GE fuel revealed that the current Technical Specification (TS) Limiting Condition for Operation (LCO) and conditions must be revised to place additional limitations on equipment to ensure that the plant is operated within the assumptions of the safety analyses. With the additional limitations, the analyses demonstrate that all of the acceptance criteria continue to be met. As a result, the changes do not involve a significant increase in the probability of consequences of an accident previously evaluated.

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

The proposed changes do not involve a physical alteration of the facility or change the manner in which the facility is operated. No new or different equipment is being installed and no installed equipment is being removed. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. Consequently, no new failure modes are introduced and the changes therefore do not increase the possibility of a new or different kind of accident from any previously evaluated.

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

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 setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes do not impact the condition or performance of structures, systems or components relied upon for accident mitigation or any safety analysis assumptions. The changes reflect a reduction in redundancy in the capability of the Automatic Depressurization System (ADS). However, the proposed changes impose more restrictive requirements on operation to ensure that all of the accident analyses continue to meet acceptance criteria. Therefore the proposed changes do not involve a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 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.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–412, Beaver Valley Power Station, Unit 2, Beaver County, Pennsylvania

Date of amendment request: March 28, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) requirements to credit the soluble boron in the fuel storage pool analyses. This amendment would revise the index, modify TS 3.9.14, ''Fuel Storage—Spent Fuel Storage Pool,'' add TS 3.9.15, ''Fuel Storage Pool Boron Concentration,'' modify applicable Bases and revise Design Feature Section 5.3.1.1, ''Criticality.'' TS 3.9.14 would be modified by separating this specification into two specifications to support crediting soluble boron in the fuel storage pool. The revised TS 3.9.14 would provide controls for fuel assembly enrichment and burnup in the spent fuel pool and also include an increase in the maximum enrichment from 4.85 weight percent (w/o) to 5.0 w/o. A new TS 3.9.15 would provide control for soluble boron requirements in the spent fuel pool. Separating this specification into two specifications follows the general guidance provided in the improved standard TS (ISTS) of NUREG–1431.

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?*

Because of the Boraflex deterioration that has been observed, the spent fuel racks have been reanalyzed neglecting the presence of Boraflex to allow storage of Westinghouse 17x17 fuel assemblies with nominal enrichments up to 5.0 weight percent (w/o) using credit for checkerboarding, burnup and soluble boron. The proposed changes will not have a significant impact on the safety of the plant or on the spent fuel storage pool and are consistent with the NRC approved changes identified for other plants (i.e., Prairie Island Units 1 and 2, Vogtle Units 1 and 2). Criteria set forth in Table 3.9–1 provide qualification requirements for fuel assembly storage to ensure the NRC acceptance criteria and accident analysis assumptions are satisfied. Increasing the enrichment from 4.85 w/o up to and including 5.0 w/o U–235 [uranium 235] has minor effects on the radiological source terms and subsequently the potential releases, both normal and accidental, are not significantly affected.

The proposed Technical Specification changes credit the use of soluble boron in the spent fuel pool criticality analyses. These criticality analyses were performed using the NRC approved methodology developed by the Westinghouse Owners Group (WOG) and described in WCAP–14416-NP-A, Revision 1, ''Westinghouse Spent Fuel Rack Criticality Analysis Methodology,'' November 1996. The analysis includes evaluations that factor in the axial burnup bias correction and utilizing identified conservatisms in the analysis demonstrate that Keff remains less than or equal to the design limits.

The proposed changes do not involve a change to plant equipment and do not affect the performance of plant equipment used to mitigate an accident. They do not affect the operation of the spent fuel pool cooling system or any other system and are consistent with applicable analyses including [those associated with postulated] fuel handling accidents. They will not affect the ability of any system to perform its design function; therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no hardware changes associated with this license amendment nor are there any changes in the method by which any safety-related plant system performs its safety function. No new accident scenarios, transient precursors, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not introduce any adverse effects or challenges to any safety-related systems.

The potential criticality accidents have been reanalyzed to demonstrate that the pool remains subcritical. Soluble boron has been maintained in the fuel storage pool water since its initial operation. The possibility of a fuel storage pool dilution is not affected by the proposed changes to the Technical Specifications. Therefore, implementation of Technical Specification controls for the soluble boron will not create the possibility of a new or different kind of accidental pool dilution.

With credit for soluble boron now a major factor in controlling subcriticality, an evaluation of fuel storage pool dilution events was completed. This evaluation concluded that no credible events would result in a reduction of the criticality margin below the 5% margin recommended by the NRC. In addition, the No Soluble Boron 95/95 probability/confidence level criticality analysis assures that dilution to 0 ppm [parts per million] will not result in criticality.

The proposed Technical Specification changes ensure the maintenance of the fuel pool boron concentration and storage configuration. Therefore, the proposed changes will not create the possibility of any 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 changes do not affect the acceptance criteria for any analyzed event nor impact any plant safety analyses since the analysis assumptions are not changed. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since the proposed changes do not affect equipment required to mitigate design basis accidents described in the Updated Final Safety Analysis Report. The Technical Specifications continue to assure that applicable operating parameters are maintained within required limits.

The proposed changes to the fuel storage pool boron concentration and storage requirements will provide adequate margin to assure that the fuel storage array will always remain subcritical by the 5% margin recommended by the NRC. These limits are based on a criticality analysis performed in accordance with NRC approved Westinghouse fuel storage rack criticality analysis methodology.

While criticality analysis utilized credit for soluble boron, the storage configurations have been defined using K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron. Soluble boron credit is used to offset off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the fuel storage pool K_{eff} is maintained less than or equal to 0.95.

The spent fuel pool boron dilution analysis concludes that an unplanned or inadvertent event which would result in dilution of the spent fuel pool boron concentration from 2000 ppm to 450 ppm is not a credible event. This conclusion is based on the substantial volume of unborated water required to dilute the pool and the fact that a large dilution event would be readily detected by plant personnel via alarms, flooding in the fuel handling building or detected during normal operator rounds through the spent fuel pool area.

The margin of safety depends upon maintenance of specific operating parameters within design limits. The Technical Specifications continue to require that these limits be maintained and provide appropriate remedial actions if a limit is exceeded. The maintenance of these limits continues to be assured through performance of surveillances. Therefore, the plant will be maintained within the analyzed limits and the proposed changes will not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: July 24, 2001.

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

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

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

The proposed change relocates several Reactor Coolant System (RCS) parameters from the ITS to the Core Operating Limits Report (COLR). The purpose for this relocation is to permit the values of these parameters to be changed under the 10 CFR 50.59 change process for cycle-specific analyses. In addition, these changes will allow increased Once-Through Steam Generator (OTSG) tube plugging. The increased plugging limit is in accordance with the analysis and will support continued proper maintenance of the OTSGs. The increased OTSG plugging will result in a small decrease in RCS flow and primary to secondary heat transfer. The difference in heat transfer results in small changes to primary and secondary operational parameters but will not result in any challenges to plant equipment. The change in RCS parameters will have no impact on the probability of accident initiators or precursors. Increased OTSG plugging will slightly reduce mass release to the containment following some loss of primary coolant accidents. Previously analyzed accidents were reevaluated considering the proposed changes and were found to be within established limits. Therefore, the change will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not introduce any new operating methods or configurations. The revised RCS parameters have been analyzed and have been determined to be within established limits. No new failure modes or limiting single failures were identified. All safety and design criteria continue to be met. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes affect RCS parameters, which are inputs to the plant's safety limits. The changes have been evaluated and the resultant plant analysis and configuration remain within the existing safety limits. The safety limits themselves are not being altered. The accident analysis was reevaluated and it has been determined that there is no significant impact on the fuel cladding, reactor coolant system or the containment structure. Therefore, this change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief, Acting: Kahtan N. Jabbour.

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: May 17, 2001.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 3/4.9.3, ''Decay Time,'' to allow the start of a core offload at 100 hours after reactor subcriticality between September 15 and June 15, and 148 hours after reactor subcriticality between June 16 and September 14. The difference in the required decay times is dependent on the time of year due to the lake temperature assumed in the spent fuel pool cooling analysis. In addition, the proposed license amendment would make format changes to the TS pages.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below: According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
2. create the possibility of a new or different kind of accident from any previously analyzed; or
3. involve a significant reduction in a margin of safety.

The determinations that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below:

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

The proposed license amendment would allow fuel assemblies to be removed from the reactor core and be stored in the spent fuel pool in less time after subcriticality than currently allowed by the TSs. Decreasing the decay time of the fuel affects the isotopic make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The proposed changes do not involve a significant increase in the probability of occurrence of an accident previously evaluated. The accident previously evaluated that is associated with the proposed license amendment is the fuel handling accident. Allowing the fuel to be offloaded as early as 100 hours after subcriticality does not impact the manner in which the fuel is offloaded. The accident initiator is the dropping of the fuel assembly. Since earlier offload does not effect fuel handling, there is no increase in the probability of occurrence of a fuel handling accident. The time frame in which the fuel assemblies are moved has been evaluated against the 10 CFR Part 20 and 10 CFR Part 100 dose limits for members of the public and licensee personnel and 10 CFR 50.67 control room dose limits. All dose limits are met with the reduced core offload times.

The proposed changes do not involve a significant increase in the consequences of an accident previously evaluated. The accident previously evaluated that is associated with fuel movement is the fuel handling accident. Thus, there is no significant increase in consequences.

The TS page format changes are administrative in nature and have no impact on any accident previously evaluated. Thus, the probability of occurrence of an accident previously evaluated is not changed.

Therefore, the proposed license amendment does not increase the probability of occurrence or the consequences of accidents previously evaluated are not increased.

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

The proposed license amendment would allow core offload to occur in less time after subcriticality, which affects the isotopic make-up of the fuel to be offloaded as well as the amount of decay heat that is present from the fuel at the time of offload. The isotopic makeup of the fuel assemblies and the amount of decay heat produced by the fuel assemblies do not currently initiate any accident. A change in the isotopic makeup of the fuel at the time of core offload or an increase in the decay heat produced by the fuel being offloaded will not cause the initiation of any accident. There is no change to the manner in which fuel is being handled or in the equipment used to offload or store the fuel. The effects of the additional decay heat load have been analyzed. The analysis demonstrated that the existing spent fuel pool cooling system and all associated systems under worst-case circumstances would maintain the integrity of the spent fuel pool and the proposed method of offload does not create a new or different kind of accident from any accident previously evaluated.

The TS page format changes are administrative in nature and have no impact on the operation of either unit. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed license amendment does 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 margin of safety pertinent to the proposed changes is the dose consequences resulting from a fuel handling accident. The shorter decay time prior to fuel movement has been evaluated against the 10 CFR Part 100 in the current licensing basis and all limits continue to be met. In addition, the integrity of the spent fuel pool has been demonstrated with the additional decay heat load. As stated above, the changes in isotopic makeup and additional heat load do not impact any safety settings and do not cause any safety limit to not be met. In addition, the integrity of the spent fuel pool is maintained.

The proposed format changes do not affect plant operation, and, therefore, do not involve a significant reduction in a margin of safety.

Therefore, the proposed changes do 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.

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

NRC Section Chief: Claudia M. Craig.

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: July 17, 2001.

Description of amendment requests: The proposed amendments would revise Technical Specification Surveillance Requirement 4.0.3 to provide a delay period following discovery of a missed surveillance prior to declaring that the Limiting Condition for Operation has not been met. The proposed delay period would be 24 hours from the time of discovery of the missed surveillance or the limit of the specified surveillance interval, whichever is less. The proposed changes are consistent with the intent of Generic Letter 87–09, ''Sections 3.0 and 4.0 of the Standard Technical Specifications on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.'' Indiana Michigan Power Company is submitting this request to reduce the potential for unnecessary plant system and equipment manipulations.

The proposed license amendment also includes format changes that improve appearance and are not intended to introduce other changes.

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

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

It is overly conservative to assume components are inoperable when a surveillance requirement has not been performed. The 24-hour delay period to perform a missed surveillance does not involve a significant increase in the probability of occurrence of an accident previously evaluated because it allows time to perform the surveillance without requiring other plant manipulations such as a plant shutdown. If a plant shutdown is required before a missed surveillance is completed, it is likely that the surveillance would be conducted when the plant is being shut down because completion of a missed surveillance would terminate the shutdown requirement. A forced plant shutdown or other forced actions prior to completion of the missed surveillance increases risk to the plant, as it requires the manipulation of additional equipment. Delaying a surveillance test on a component cannot cause a failure of the component, nor would it significantly affect accident initiators or precursors. Therefore, there is no significant increase in the probability of occurrence of an accident previously evaluated.

Since this change does not affect plant design, operation, or the manner in which testing is performed, there is no effect on the consequences of an accident previously evaluated.

The T/S page format changes are administrative in nature and have no impact on plant operation.

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

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

The proposed change does not affect plant design, operation, or the manner in which testing is performed. Delaying a surveillance test on a component cannot cause a failure of the component. As such, the proposed delay period will not cause any equipment malfunctions or introduce any changes to the way in which components operate. The T/S page format changes are administrative in nature and have no impact on plant operation. Therefore, the proposed changes do not increase 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 margin of safety is neither described or prescribed for this specification. The proposed change simply provides additional time to perform a surveillance and verify that the operability of equipment is in conformance with the T/S requirements.

The T/S page format changes are administrative in nature and have no impact on plant operation.

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 requests involve no significant hazards consideration.

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment requests: June 29, 2001.

Description of amendment requests: The licensee requests to revise Technical Specifications (TSs) 3.7.10, ''Emergency Chilled Water (ECW)'' and 3.7.11, ''Control Room Emergency Air Cleanup System (CREACUS)'' and the associated TSs Bases. The proposed change would revise the Allowed Outage Time (AOT) for a single inoperable train of both the ECW and CREACUS from 7 days to 14 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:

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a 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. A discussion of these standards as they relate to this amendment request follows:

(1) *Will 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: No.

This proposed change is to revise the Allowed Outage Time (AOT) for a single inoperable train of the Emergency Chilled Water (ECW) and Control Room Emergency Air Cleanup System (CREACUS) systems from 7 days to 14 days. The proposed change does not involve a change in the design configuration, or operation of the plant.

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

(2) *Will 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: No.

This proposed change does not involve a change in the design, configuration, or method of operation of the plant.

Therefore, this proposed change will not create the possibility of a new or different kind of accident from any accident that has been previously evaluated.

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

Response: No.

The proposed change does not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. Probabilistic risk analysis was used to evaluate these changes.

Therefore, there will be no significant reduction in a margin of safety as a result of this change.

Based on the responses to these three criteria, Southern California Edison (SCE) 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: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 9, 2001.

Description of amendment request: Proposed amendments revise Technical Specification 3.5.1, ''Emergency Core Cooling Systems—Accumulators,'' to extend the allowed outage time allowed for an inoperable accumulator to 24 hours.

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:

STPNOC 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,'' as discussed below.

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

Response: No.

The proposed changes involve no significant increase in the probability of an accident previously evaluated because the accumulator has no role as an accident initiator.

The proposed extension to the allowed outage time has no significant effect on the availability of the accumulator to perform its design function and has no effect on the configuration or accident response of the accumulator. The proposed change involves no changes to the accident analyses. Consequently, the proposed extended allowed outage time involves no significant increase in the consequences of an accident previously evaluated.

The proposed changes to eliminate the surveillance requirements also have no significant effect on the availability of the accumulator to perform its design function and have no effect on the configuration or accident response of the accumulator. The changes to the surveillance requirements involve no change to the accident analyses. Consequently, the changes to the surveillance requirements involve no significant increase in the consequences of an accident previously evaluated.

The proposed changes in the structure of the specification to be more consistent with ITS are administrative and have no technical impact. Consequently, they involve no significant increase in the probability or consequences of an accident previously evaluated.

The correction of the typographical error is an administrative change which has no operational significance.

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

Response: No.

The proposed change does not involve the installation or operation of any new or different kinds of equipment, nor does it involve a new or different mode of operation. The proposed changes do not result in systems operating in a manner different from existing procedures and practices. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes in the structure of the specification to be more consistent with ITS are administrative and have no technical impact. Consequently, they do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The correction of the typographical error is an administrative change which has no operational significance.

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

Response: No.

The proposed change will allow plant operation in a configuration outside the design basis for up to 24 hours before being required to begin shutdown. The impact of this on plant risk was evaluated and found to be very small. That is, increasing the time the accumulators will be unavailable to respond to large LOCA event, assuming design basis accumulator success criteria is necessary to mitigate the event, has a very small impact on plant risk. The analyses quantitatively demonstrate the change does not involve a significant reduction in the margin of safety.

The proposed change removes the 18 month test to verify that the accumulator isolation valves automatically open when a simulated or actual P–11 interlock setpoint is exceeded, or when an SI signal is received. The valves are verified open every 24 hours and the power is verified removed every 31 days in accordance with the TS. Should the valves be inadvertently closed, the normal testing would adequately identify the condition. If the condition is recognized, the failure would be addressed by plant administrative controls that would immediately result in the appropriate Actions being taken for all affected systems. Based on the existence of other measures which adequately address the reason for the current requirement, this change does not involve a significant reduction in a margin of safety.

The proposed change removes the requirement from the Technical Specifications to perform surveillances on the accumulator instrumentation. The TS does not specifically require this instrumentation to be used to meet the required pressure and level verification surveillances. The verification of accumulator level and pressure may be determined by either installed instrumentation or temporary test equipment. Therefore, the change does not involve a significant reduction in a margin of safety.

The proposed changes in the structure of the specification to be more consistent with ITS are administrative and have no technical impact. Consequently, they do not involve a significant reduction in the margin of safety.

The correction of the typographical error is an administrative change which has no operational significance.

Based upon the analysis provided herein, the proposed amendments will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident previously evaluated, or involve a reduction in a margin of safety. Therefore, the proposed amendments meet the requirements of 10 CFR 50.92 and do not involve a significant hazards consideration.

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

Attorney for licensee: Alvin H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: May 30, 2001.

Description of amendment request: Proposed amendments would permit relaxation of the allowed outage times and bypass test times for limiting conditions for operations under Technical Specifications 3.31, ''Reactor Trip System Instrumentation,'' and 3.3.2, ''Engineered Safety Features Actuation System Instrumentation.''

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

Response: No.

The reactor protection and engineered safety features functions are not initiators of any design basis accident or event and therefore the proposed changes do not increase the probability of any accident previously evaluated. The proposed changes to the allowed outage and bypass test times have an insignificant impact on plant safety based on the calculated core damage frequency increase being approximately 1.OE–06. Therefore, the proposed changes do not result in a significant increase in the consequences of an accident previously evaluated.

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

Response: No.

The proposed changes do not result in a change in the manner in which the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) provide plant protection. The existing RTS and ESFAS actuation setpoints will be unaffected by these proposed changes. The changes to the allowed outage and bypass test times do not change any existing accident scenarios nor create any new or different accident scenarios. Therefore, this request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The impact of increased allowed outage times and bypass test times should result in an overall improvement in safety by reducing the potential for spurious reactor trips and spurious actuation of safety equipment. The longer allowed outage times and bypass test times will provide additional time before being required to place the associated channel in trip. With the channel in trip, the logic required to cause a reactor trip or safety system actuation is reduced to 1-out-or-2 (for 2-out-of-3 logic) and 1-out-of-3 (for 2-out-or-4 logic). With one channel tripped, the potential for a spurious actuation is increased. Placing a channel in bypass for additional time does reduce the availability of signals to initiate component actuation for event mitigation when required, but as shown in WCAP–14333, the impact on safety is small due to the availability of other signals or operator action to trip the reactor or cause component actuation. Therefore, these proposed changes should reduce the potential for inadvertent reactor trips and inadvertent equipment actuations due to human error or spurious actuation, and will not involve a significant reduction in the margin of safety.

Based upon the analysis provided herein, the proposed amendments will not increase the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a reduction in a margin of safety. Therefore, the proposed amendments meet the requirements of 10 CFR 50.92 and do not involve a significant hazards consideration.

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

Attorney for licensee: Alvin H. Gutterman, Esq., Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 6, 2001 (TS 01–05).

Description of amendment request: The proposed amendment would revise the Technical Specification (TS) surveillance requirements for containment isolation valves (CIVs) to be verified closed. More specifically, valves in high radiation areas may be verified by administrative means. In addition, valves which are locked sealed or otherwise secured do not need to be reverified closed and are eliminated from the scope of the surveillance.

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

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

The proposed changes to the surveillance requirements (SR) for verification of valve position continues to assure the operability of these valves such that the containment isolation function assumed in the safety analyses is maintained. Since these proposed revisions will continue to support the required safety functions without modification of the plant features, the probability of an accident is not increased.

The provisions proposed in this change request will continue to maintain an acceptable level of protection for the health and safety of the public and will not impact the potential for the offsite release of radioactive products. The overall effect of the proposed change will result in specifications that have equivalent requirements compared to existing specifications for CIV operability and will not increase the consequences of an accident.

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

The proposed revisions are not the result of changes to plant equipment, system design, testing methods, or operating practices. The modified requirements will allow the use of administrative means for verification of valve closure for those CIVs located in high radiation areas and eliminate the requirement to verify close those valves that are locked, sealed, or otherwise secured. The specifications for CIVs serve to provide controls for maintaining the containment pressure boundary. TVA's proposed changes does not contribute to the generation of postulated accidents. Since the function of the CIVs and their associated systems remains unchanged, and the effects do not contribute to accident generation, the proposed changes will not create the possibility of a new or different kind of accident.

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

The proposed change involves upgrading the CIV TS surveillance requirement to be consistent with the S[Standard]TS. The proposed change has been developed considering the importance of the CIVs in limiting the consequences of a design basis event and the concerns for the plant's ability to perform required operational support functions with the necessary systems isolated. The proposed change allows for alternative protection to assure the isolation function of the valves remain available.

Since the proposed revision does not alter the intent or application of the current TS requirements, and the function of the CIVs and their associated systems remains unchanged, the proposed change will continue to provide controls for maintaining the containment pressure boundary.

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

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

NRC Section Chief: Richard P. Correia.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: June 5, 2000, as supplemented August 4, 2000, and July 6, 2001.

Brief description of amendment: This amendment revises Technical Specification (TS) 3.7.8 to establish Required Actions and Completion Times in the event that the service water system exceeds the maximum allowed TS temperature of 97 degrees F.

Date of issuance: August 9, 2001.

Effective date: August 9, 2001.

Amendment No. 191.

Facility Operating License No. DPR–23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** August 9, 2000 (65 FR 48745). The August 4, 2000, and July 6, 2001, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 9, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 18, 2001

Brief description of amendment: The amendment revises Technical Specification (TS) 3/4.9.4 ''Containment Building Penetrations'' and the associated Bases to permit containment building penetrations to remain open, under administrative controls, during core alterations or the movement of irradiated fuel within the containment. Specifically, the amendment: (1) Incorporates an alternate source term methodology in the fuel handling accident analysis; (2) revises TS 3.9.4 to remove portions of a note restricting the applicability of administrative controls with respect to containment penetrations; and (3) includes the use of administrative controls on the equipment hatch and other penetrations that provide access from containment atmosphere to outside atmosphere.

Date of issuance: July 30, 2001.

Effective date: July 30, 2001.

Amendment No.: 104.

Facility Operating License No. NPF–63: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34280).

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

No significant hazards consideration comments received: No.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: May 10, 2001:

Brief description of amendment: The amendment removes Technical Specification surveillance requirement 4.6.A.4 that requires each emergency diesel generator (EDG) to be given a thorough inspection at least annually following the manufacturer's recommendations. The requirement for the EDG inspection will be relocated to the Updated Final Safety Analysis Report and will be in accordance with the licensee-controlled maintenance program. The inspection period required by the maintenance program will also be changed to specify that it will be ''in accordance with the manufacturer's recommendations.''

Date of issuance: July 30, 2001.

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

Amendment No.: 218.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31704).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 1, 2001.

Brief description of amendments: The amendments revised the Technical Specifications (TS) to permit implementation of 10 CFR part 50, Appendix J, Option B and to reference Regulatory Guide 1.163, ''Performance-Based Containment Leak Test Program,'' dated September 1995, which specifies a method acceptable to the NRC for complying with Option B. These changes relate only to Type B and Type C (local) leakage rate testing. In addition, the amendments revised Surveillance Requirement 3.6.3.8 by deleting the requirement for soap bubble testing of welded penetrations that are not individually testable and clarified the Bases for TS 3.6.2 pertaining to the containment air lock door.

Date of issuance: July 31, 2001.

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

Amendment Nos.: 192/184.

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

Date of initial notice in **Federal Register:** April 19, 2001 (66 FR 22028).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 31, 2001.

No significant hazards consideration comments received: No.

Entergy Nuclear Operations, Inc., Docket No. 50–333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: May 11, 2001.

Brief description of amendment: The amendment extends, on a one-time-basis, the Limiting Condition for Operation allowable out-of-service time for the residual heat removal service water (RHRSW) system from 7 days to 11 days. The applicability of this change is limited to the one-time-only installation of the modification to the ''B'' RHRSW strainer.

Date of issuance: July 27, 2001.

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

Amendment No.: 271.

Facility Operating License No. DPR–59: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34282).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: February 9, 2001 as supplemented by letters dated May 18, 2001 and June 26, 2001.

Brief description of amendments: The one-time amendments revise Braidwood Unit 1 Technical Specifications (TS), section 5.5.9.d.2, ''Steam Generator Tube Surveillance Program, Inspection Frequencies,'' for the Braidwood Station, Unit 1, fall 2001 refueling outage to allow a 40 month inspection interval after its first (post-replacement) inservice inspection, resulting in a C–1 classification, rather than after two consecutive inspections.

Date of issuance: August 9, 2001.

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

Amendment Nos.: 117 and 117.

Facility Operating License Nos. NPF–72 and NPF–77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22030). The May 18, 2001 and June 26, 2001, supplemental letters provided clarifying information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 9, 2001.

No significant hazards consideration comments received: No.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50–334, Beaver Valley Power Station, Unit No. 1 (BVPS–1), Beaver County, Pennsylvania

Date of application for amendment: December 21, 2000.

Brief description of amendment: The amendment revised Technical Specification 3/4.3.1, ''Reactor Trip System Instrumentation,'' and associated bases to reflect the deletion of the steam/feedwater flow mismatch and low steam generator water level reactor trip function.

Date of issuance: August 8, 2001.

Effective date: As of the day of issuance and shall be implemented by the first entry into MODE 2 following the BVPS–1 Refueling Outage 14.

Amendment No: 240.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 24, 2001 (66 FR 7680).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: December 27, 2000, as supplemented on March 28, April 12, June 9, June 13, and June 29 (3), 2001. The addition of a Technical Specification (TS) Bases control program was requested on March 28, 2001.

Brief description of amendments: These amendments allow: (1) Revisions to reactor trip and engineered safety feature actuation setpoints and allowable values, (2) implementation of the revised thermal design procedure, (3) relocations of TS requirements to the core operating limits report, (4) relocation of TS requirements to the licensee requirements manual, (5) miscellaneous editorial changes. In addition, License Condition 2.(C).(3) regarding less than 3-loop operation was deleted.

Date of issuance: July 20, 2001.

Effective date: Immediately and to be implemented within 120 days.

Amendment Nos.: 239 and 120.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications and License.

Date of initial notice in **Federal Register:** April 18, 2001 (66 FR 20002) for the December 27, 2000, amendment request. A portion of a March 28, 2001, amendment request was also issued in this amendment. The date of the initial notice for the March 28, 2001, amendment request was June 20, 2001 (66 FR 33111).

The March 28, April 12, June 9, June 13, and June 29 (3), 2001, letters provided clarifying information that did not change the initial proposed 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 July 20, 2001.

No significant hazards consideration comments received: No.

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 application for amendments: March 12, 2001 as supplemented June 26, 2001.

Brief description of amendments: The amendments to the emergency diesel generators (EDG) Technical Specifications (TS) revised the 72-hour allowed outage time specified in TS 3.8.1.1, Actions b and f, and Tss 3.4.3 and 3.5.2 to allow 14 days to restore an inoperable EDG to operable status. In addition, the amendments deleted TS Surveillance Requirement 4.8.1.1.2.g.1 and allowed its relocation to a licensee-controlled maintenance program that will be incorporated by reference into the Updated Final Safety Analysis Report.

Date of issuance: August 8, 2001.

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

Amendment Nos: 215 and 209.

Facility Operating License Nos. DPR–31 and DPR–41: Amendments revised the TS.

Date of initial notice in **Federal Register:** April 18, 2001 (66 FR 20005). The supplemental submittal of June 26, 2001, provided clarifying information that did not change the scope of the original request or change the initial proposed no significant hazards consideration.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 8, 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: August 18, 2000.

Brief description of amendments: The amendments would change Technical Specification (TS) 3/4.7.4, ''Essential Service Water (ESW) System,'' and the associated Bases to add requirements that would support cross-connection to the opposite unit. The proposed amendment would also delete a provision for a 60-day allowed outage time when an ESW flowpath is not available to support the opposite unit's shutdown functions. Administrative and editorial changes are also made to provide consistency between units, correct typographical errors, improve readability, and improve page layout.

Date of issuance: August 3, 2001.

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

Amendment Nos.: 253 and 235.

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

Date of initial notice in **Federal Register:** September 20, 2000 (65 FR 56951) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 3, 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: July 17, 2001.

Brief description of amendments: The amendments revise Technical Specification 3.3.1.1, Table 3.3–1, Action 2a, to increase the amount of time allowed to place an inoperable power range neutron flux channel in the tripped condition from one hour to six hours.

Date of issuance: August 8, 2001.

Effective date: As of the date of issuance and shall be implemented within 3 days.

Amendment Nos.: 254 and 236.

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

Public comments requested as to proposed no significant hazards consideration: Yes (66 FR 38753, dated July 25, 2001). The notice provided an opportunity to submit comments on the Commission's proposed NSHC determination. No comments have been received. The notice also provided for an opportunity to request a hearing by August 24, 2001, but indicated that if the Commission makes a final NSHC determination, any such hearing would take place after issuance of the amendment.

The Commission's related evaluation of the amendments, finding of exigent circumstances, state consultation, and final NSHC determination are contained in a Safety Evaluation dated August 8, 2001.

Attorney for licensee: David W. Jenkins, Esq., Indiana Michigan Power Company, Nuclear Generation Group, One Cook Place, Bridgman, MI 49106.

NRC Section Chief: Claudia M. Craig.

Niagara Mohawk Power Corporation, Docket No. 50–220, Nine Mile Point Nuclear Station Unit No. 1, Oswego County, New York

Date of application for amendment: March 6, 2001.

Brief description of amendment: The amendment revises the Technical Specifications to change the standard by which the licensee tests charcoal used in engineered safeguards features systems to American Society for Testing and Materials D3803–1989. These revisions are made in accordance with Generic Letter 99–02, ''Laboratory Testing of Nuclear-Grade Activated Charcoal.''

Date of issuance: July 30, 2001.

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

Amendment No.: 171.

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

Date of initial notice in ***Federal Register***: April 4, 2001 (66 FR 17968).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated

No significant hazards consideration comments received: No.

Niagara Mohawk Power Corporation, Docket No. 50–410, Nine Mile Point Nuclear Station Unit No. 2, Oswego County, New York

Date of application for amendment: March 29, 2001.

Brief description of amendment: The amendment revises the Technical Specifications, Section 3.7.2, ''Control Room Envelope Filtration (CREF) System,'' to establish actions to be taken for an inoperable CREF system due to a degraded control room envelope boundary.

Date of issuance: August 7, 2001.

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

Amendment No.: 97.

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

Date of initial notice in ***Federal Register***: May 30, 2001 (66 FR 29360).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 19, 2000, as supplemented March 23, April 9, and June 27, 2001.

Brief description of amendment: The amendment revises the licensing basis to utilize the full scope of an alternative radiological source term for accidents as described in NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' and revises the Technical Specifications implementing various assumptions in the alternative source term analyses.

Date of issuance: July 31, 2001.

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

Amendment No.: 240.

Facility Operating License No. DPR–49: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: March 6, 2001 (66 FR 13598).

The March 6, 2001, notice provided an opportunity for a hearing and petition for leave to intervene. No requests for hearing or petition for leave to intervene were received. Subsequently, the staff determined that the licensing action was eligible for categorical exclusion from environmental review. The amendment request was noticed on June 27, 2001

(66 FR 34285), with the staff's proposed no significant hazards consideration determination. The June 27, 2001, supplement contained corrected TS pages 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 July 31, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 2, 2001, as supplemented June 22 and July 27, 2001.

Brief description of amendment: The amendment (1) relocates requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section XI, Inservice Testing (IST) Program currently contained in Technical Specification (TS) Surveillance Requirement (TSSR) 4.15.B to TS Administrative Control Section 6.8, ''Programs and Manuals,'' (2) makes conforming changes to several SRs to reflect the change in reference from TSSR 4.15.B to the licensee-controlled IST Program, (3) rewords TSSRs 4.5.A.3 and 4.5.D.1 to be consistent with NUREG–1433, (4) incorporates TS Task Force (TSTF) initiative TSTF–279 into TS Administrative Control Section 6.8, and (5) revises TSSRs 4.6.H.1, 4.6.H.3, and Table 4.6.1 to change the inspection and functional testing interval extensions reference from plus-or-minus 25 percent to plus 25 percent.

Date of issuance: August 1, 2001.

Effective date: As of the date of issuance and shall be implemented within 45 days.

Amendment No.: 122.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register***: May 30, 2001 (66 FR 29360).

The June 22, 2001, supplement provided clarifying information to the application and added a table defining IST testing frequencies to the proposed TS 6.8.G in order to be consistent with NUREG–1433. The July 27, 2001, supplement provided updated TS pages to reflect amendments issued subsequent to the application. The supplements were within the scope of the original **Federal Register** notice and did not change the staff's initial proposed no significant hazards considerations determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 2, 2001.

Brief description of amendment: Removes from the Technical Specifications all requirements for, and references to, the term ''Assembly Radial Peaking Factor.''

Date of issuance: August 1, 2001.

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

Amendment No.: 205.

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

Date of initial notice in ***Federal Register:*** May 2, 2001 (66 FR 22027).

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

No significant hazards consideration comments received: No.

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

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

Brief description of amendment: The amendment revises Section 5.0, ''Administrative Controls,'' of the Permanently Defueled Technical Specifications by eliminating the position of Senior Vice President, Power Supply, and assigning those duties to the Trojan Site Executive; and dividing the position and duties of the Trojan Site Executive and Plant General Manager between two separate positions: (1) Trojan Site Executive, and (2) General Manager, Trojan. The amendment also revises the language used in Section 5.0 of the Permanently Defueled Technical Specifications to conform with the language of revised 10 CFR 50.59 by replacing phrases which included the wording ''unreviewed safety question'' and ''safety evaluation'' with wording that will continue to conform to the requirements of revised 10 CFR 50.59.

Date of issuance: July 31, 2001.

Effective date: This license amendment is effective as of the date of issuance.

Amendment No.: 207.

Facility Operating License No. NPF–1: The amendment changes the Permanently Defueled Technical Specifications.

Date of initial notice in **Federal Register:** April 4, 2001 (66 FR 17962).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 31, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 14, 2000, as supplemented April 24 and May 24, 2001.

Brief description of amendment: The amendment removes the references to the Independent Safety Engineering Group.

Date of issuance: July 30, 2001.

Effective date: July 30, 2001.

Amendment No.: 151.

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

Date of initial notice in **Federal Register:** November 1, 2000 (65 FR 65349). The April 24 and May 24, 2001, supplements contained clarifying information only and did not change the initial no significant hazards consideration determination or expand the scope of the initial application.

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

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: May 30, 2001 (ULNRC–04481).

Brief description of amendment: The amendment removes the phrase ''and the charging flow control valve full open'' from Limiting Condition for Operation 3.5.5, Required Action A.1, and Surveillance Requirement 3.5.5.1 for the reactor coolant pump seal injection flow in the technical specifications.

Date of issuance: August 7, 2001.

Effective date: August 7, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 146.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34289)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2001.

No significant hazards consideration comments received: No.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: March 22, 2001.

Brief description of amendment: The amendment changed the penetration values in Technical Specification (TS) 5.5.11.c for laboratory testing of the charcoal adsorber for the control room ventilation system from 2 percent to 2.5 percent and the auxiliary/fuel building emergency exhaust system from 2 percent to 5 percent. The amendment also deleted the ''≤'' sign associated with the temperature for the laboratory test of a sample of the charcoal adsorber.

Date of issuance: August 7, 2001.

Effective date: August 7, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 139.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 16, 2001 (66 FR 27178).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 7, 2001.

No Significant Hazards Consideration comments received: No.

Dated at Rockville, Maryland, this 14th day of August 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–20885 Filed 8–21–01; 8:45 am]

BILLING CODE 7590–01–P

PRESIDIO TRUST

Notice of Receipt of and Availability for Public Comment on an Application for Wireless Telecommunications Facilities Site; The Presidio of San Francisco, California

AGENCY: The Presidio Trust.

ACTION: Public notice.

SUMMARY: This notice announces the Presidio Trust's receipt of and availability for public comment on an application from GTE Mobilnet of California d/b/a Verizon Wireless for colocation at an existing wireless telecommunications facilities site (''Project'') in The Presidio of San Francisco. The proposed location of the Project is in the vicinity of 1255 Armistead Road.

The Project involves (i) replacing an existing utility pole (installed by AT&T Wireless) with a taller pole to accommodate two additional antenna panels, and (ii) placing the associated radio equipment within a new prefabricated equipment shelter. The utility pole will be approximately 60 feet tall, 10 feet taller than the existing AT&T Wireless pole. Power for the project will be provided through underground coaxial cables connected to existing power sources. Connection to telephone lines will be through existing telephone lines.

COMMENTS: Comments on the proposed project must be sent to Celeste Evans, Presidio Trust, 34 Graham Street, P.O. Box 29052, San Francisco, CA 94129–0052, and be received by September 24, 2001. A copy of Verizon's application is available upon request to the Presidio Trust.

FOR FURTHER INFORMATION CONTACT: Celeste Evans, Presidio Trust, 34 Graham Street, P.O. Box 29052, San Francisco, CA 94129–0052. Email: cevans@presidiotrust.gov. Telephone: 415–561–5300.

Dated: August 16, 2001.

Karen A. Cook,

General Counsel.

[FR Doc. 01–21139 Filed 8–21–01; 8:45 am]

BILLING CODE 4310–4R–U

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. Chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposal(s)

(1) *Collection title:* Earnings Information Request.

(2) *Form(s) submitted:* G–19–F.

(3) *OMB Number:* 3220–0184.

(4) *Expiration date of current OMB clearance:* 10/30/2001.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Individuals or Households.

(7) *Estimated annual number of respondents:* 1,500.

(8) *Total annual responses:* 1,500.

(9) *Total annual reporting hours:* 200.

(10) *Collection description:* Under Section 2 of the Railroad Retirement Act, an annuity is not payable or is reduced by any month(s) in which the beneficiary works for a railroad or earns more than the prescribed amounts. The collection obtains earnings information not previously or erroneously reported by a beneficiary.

Additional Information or Comments

Copies of the forms and supporting documents can be obtained from Chuck Mierzwa, the agency clearance officer (312–751–3363). Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement Board, 844 North Rush Street, Chicago, Illinois, 60611–2092 and the OMB reviewer, Marcie Brown (202–395–7316), Office of Management and Budget, Room 10230, New Executive Office Building, Washington, DC 20503.

Chuck Mierzwa,

Clearance Officer.

[FR Doc. 01–21108 Filed 8–21–01; 8:45 am]

BILLING CODE 7905–01–M

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

Upon Written Request, Copies Available From: Securities and Exchange Commission, Office of Filings and Information Services, Washington, DC 20549: Extension: Rule 11Ac1–4, SEC File No. 270–405, OMB Control No. 3235–0462.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*) the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget a request for extension of the previously approved collection of information discussed below.

Rule 11Ac1–4 [17 CFR 240.11Ac1–4] under the Securities Exchange Act of 1034 requires specialists and market makers to publicly display a customer limit order when that limit order is priced superior to the quote that is currently being displayed by the specialist or market maker. Customer limit orders that match the bid or offer being displayed by the specialist or market maker must also be displayed if the limit order price matches the national best bid or offer. It is estimated that approximately 926 broker and dealer respondents incur an aggregate burden of 9,056 hours per year to comply with this rule.

Rule 11Ac1–4 does not contain record retention requirements. Compliance with the rule is mandatory. Responses