2001, (66 FR 42895). However, by letter dated September 21, 2001, the licensee withdrew the proposed change.

For further details with respect to this action, see the application for amendment dated August 2, 2001, as supplemented by their letters dated August 6, 2001, August 7, 2001, and the licensee's letter dated September 21, 2001, which withdrew the application for license amendment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville,Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/NRC/ADAMS/index/html.* 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.*

Dated at Rockville, Maryland, this 11th day of October 2001.

For the Nuclear Regulatory Commission.

Mahesh Chawla,

Project Manager, Section 2, Project Directorate III,Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 01–26106 Filed 10–16–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Twenty-Ninth Nuclear Safety Research Conference

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of meeting.

SUMMARY: The Twenty-Ninth Nuclear Safety Research Conference (NSRC), formerly known as the Water Reactor Safety Meeting, will be held October 22–24, 2001, from 8 a.m. to 5 p.m. at the Marriott Hotel at Metro Center, 775 12th Street, NW., Washington, DC.

Please note that while the name of the conference has changed to more accurately reflect the broad range of topics that we now cover, the objective is still to promote dialogue with stakeholders about research that develops and confirms technical bases for regulatory decisions and prepares the Agency for the future.

Ashok C. Thadani, Director of the Office of Nuclear Regulatory Research, will open the conference on Monday, October 22, 2001, and NRC Chairman Richard Meserve will be the keynote speaker. Roy Zimmerman, Deputy Director of the Office of Nuclear Regulatory Research, will follow Chairman Meserve by discussing recent accomplishments in the Office of Nuclear Regulatory Research.

An expert panel will provide an overview of safety research programs worldwide. Panel members will include Dr. Michel Livolant, Institute de Protection et de Surete Nucleaire of France; Dr. William Magwood, U.S. Department of Energy; Dr. Theodore Marston, Electric Power Research Institute; Dr. Kunihisa Soda, Japan Atomic Energy Research Institute; and Dr. Ashok Thadani, NRC.

NRC Commissioner Greta J. Dicus will be a guest speaker at the Monday morning plenary session.

Technical sessions on advanced reactors and dry cask research will be held in the afternoon.

On Tuesday, October 23, 2001, two expert panel sessions are planned in the morning. The first expert panel on waste and decommissioning will start at 8 a.m. and will discuss current research initiatives for addressing issues in human and environmental health risk assessment. Panel members will include NRC Commissioner Edward McGaffigan, Jr; Mr. Andrew Wallo, U.S. Department of Energy; Mr. Michael Boyd, U.S. Environmental Protection Agency; Mr. Thomas Cardwell, Texas Department of Health; and Mr. Luc Baekelandt, Federal Agency for Nuclear Control in Belgium.

The other expert panel will be on advanced reactors and will provide an overview of ongoing programs and a discussion of the safety attributes of advanced designs, key issues in licensing and development, research needs and priorities, and the outlook for the future. Panel members will include NRC Commissioner Jeffrey S. Merrifield; Dr. Ron Simard, Nuclear Energy Institute; Dr. Theodore Marston, Electric Power Research Institute; Dr. William Magwood, U.S. Department of Energy; Dr. Vladimir Asmolov, Kurchatov Institute of Russia; Mr. Peter Lyons, U.S. Senate Staff (Senator Peter Domenici); and Mr. Edward Lyman, Nuclear Control Institute.

Technical sessions on fuels research and age-related issues and research will be held in the afternoon.

On Wednesday, October 24, 2001, NRC Commissioner Nils J. Diaz will provide brief remarks at 8 a.m. and will be followed by two expert panels and two technical sessions are planned. The first panel will start at 8:15 a.m. and will explore and seek innovative ways to communicate the role, scope, and content of the Office of Nuclear Regulatory Research program. Panelists include Mr. Dwight Cates, Committee on Energy and Commerce, U.S. House of Representatives; Ms. Maureen Conley of Inside N.R.C.; Ms. Angie Howard, NuclearEnergy Institute; Professor Andrew Kadak, Massachusetts Institute of Technology; Mr. David Lochbaum, Union of Concerned Scientists; Dr. Timo Okkonen, STUK—Radiation and Nuclear Safety Authority; Ms. Margaret Federline, USNRC; and Ms. Patricia Norry, Deputy Executive Director for Management Services, USNRC.

The second expert panel will be on fuels. It will look at issues to be addressed in an NRC safety research program and discuss whether the current spectrum of research projects are adequate.

Technical sessions on fuels and risk-informing regulatory practices will be held for the remainder of the day.

This international conference includes presentations by personnel from the U.S. Government, national laboratories, private contractors, universities, reactor vendors, and a number of foreign organizations.

Those who wish to attend are encouraged to register in advance on the NSRC website (*www.bnl.gov/NSRC*) or by contacting Susan Monteleone, Brookhaven National Laboratory, Department of Nuclear Energy, Building 130, Upton, NY 11973, telephone (631) 344–7235; or Sandra Nesmith (301) 415–6437, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Dated at Rockville, Maryland, this 2nd day of October 2001.

For the Nuclear Regulatory Commission.

Mabel F. Lee,

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

[FR Doc. 01–26107 Filed 10–16–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 24, 2001 through October 5, 2001. The last biweekly notice was published on October 3, 2001 (66 FR 50463).

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

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 16, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Branch, or may be delivered to the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

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

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

Date of amendment request: August 13, 2001.

Description of amendment request: The proposed amendment would change the requirement to withdraw the first set of reactor vessel surveillance specimens by deferring withdrawal for one additional operating cycle.

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

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

The withdrawal in Fall 2003 refueling outage vice the March 2002 refueling outage and the deferral of the withdrawal of the vessel surveillance specimens are not initiators of or precursors to any of the accident scenarios presented in the [Updated Safety Analysis Report] USAR. This schedular adjustment will not increase the likelihood of equipment failure, will not defeat the design reactor protection functions, and will not increase the likelihood of a catastrophic failure of any plant structure, system, or component. The vessel surveillance specimens are used as the basis for the pressure-temperature (P/T) curves. However, despite the deferral for one cycle of withdrawal of the vessel surveillance specimens, the P/T curves will continue to conservatively be established in accordance with Regulatory Guide (RG) 1.99, ''Radiation Embrittlement of Reactor Vessel Materials,'' Revision 2, as described in the USAR. Therefore, this change does not involve an increase in the probability of any accident previously evaluated.

The proposed change to the withdrawal schedule for the vessel surveillance specimens postpones the collection of one of two sets of data needed to confirm the basis of the P/T curves with no change to the currently allowed P/T curves. The P/T curves that are in the [Technical Specifications] TS will continue to be based on RG 1.99. The deferral of the removal of the first set of specimens will not affect the confirmation of the bases for the P/T curves because the withdrawal schedule for the second set of specimens is not being changed with this request. Because the basis for the P/T curves is maintained, this proposed change does not impact or increase the assumed radionuclide source term and will not result in an unacceptable reduction in reactor vessel toughness. Therefore, this change does not involve an increase in the consequences of any accident previously evaluated.

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

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

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

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

The deferral for one cycle of the withdrawal of the vessel surveillance specimens does not affect the P/T curves, and therefore does not affect the margin to safety for brittle fracture. Because two sets of specimens are needed to confirm the basis for the P/T temperatures and because the schedule for the withdrawal of the second set of specimens is not changing, the P/T curves continue in the interim to conform to RG 1.99. The proposed change does not challenge the integrity of the fuel cladding, reactor coolant pressure boundary that includes the reactor vessel, or the primary containment.

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

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

Attorney for licensee: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515

NRC Section Chief: Anthony J. Mendiola

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

Date of amendment request: September 17, 2001.

Description of amendment request: The proposed amendment would modify the Technical Specification (TS) surveillance requirement for the containment spray nozzles by changing the test frequency from ''once per 10 years'' to ''following activities that could result in nozzle blockage.''

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

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

The proposed change revises the testing requirements for the containment spray nozzles to only require verification that each spray nozzle is unobstructed following activities that could result in nozzle blockage. The only event for which the containment spray system is considered an initiator is the maximum containment negative pressure event. This event involves inadvertent actuation of containment spray following a break in the reactor water cleanup system inside containment described in Updated Safety Analysis Report (USAR) Section 6.2.1.1.4.2. This change does not increase the likelihood for an inadvertent actuation of the containment spray system.

The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. No active or passive failure mechanisms that could lead to an accident are affected. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. As a result, the probability of any accident previously evaluated, is not significantly increased.

The consequences of a previously evaluated accident are not significantly increased. The proposed change revises the current Surveillance Frequency from 10 years to following activities that could result in spray nozzle blockage. Since activities that could introduce foreign material into the system (such as inadvertent actuation of the containment spray system or loss of foreign material control) are the most likely cause for obstruction, testing or inspection following such activities would verify the nozzle(s) being unobstructed, and the system capable of performing its safety function. No other evolutions require the system boundary to be breached, so introduction of debris during times when maintenance activities are not in progress are precluded. Introduction of foreign materials into the system from the exterior is highly unlikely due to the location of the spray headers, the passive nature of the nozzles, and the fact that the containment spray headers are maintained dry which does not lend itself to active degradation mechanisms such as corrosion. The proposed testing requirements are considered sufficient to provide a high degree of confidence that containment spray flow will be available when required. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

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

The proposed change to the test frequency for the containment spray system nozzles does not involve the use or installation of new equipment. Installed equipment is not operated in a new or different manner. No new or different system interactions are created, and no new processes are introduced. The current foreign material exclusion practices have been reviewed and judged sufficient to provide high confidence that debris will not be introduced during times when the system boundary is breached.

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

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

The revision to the containment spray nozzle testing frequency does not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the containment spray system is unchanged. Since the system is not susceptible to corrosion induced obstruction nor is the introduction of foreign material from the exterior likely, the proposed testing frequency is sufficient to provide high confidence that the containment spray system will be available to provide the flow necessary to ensure that the effects of drywell bypass leakage and low energy line breaks are mitigated. Therefore, the capacity of the system will remain unchanged. As a result, this 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: Robert Helfrich, Mid-West Regional Operating Group, Exelon Generation Company, LLC, 1400 Opus Place, Suite 900, Downers Grove, IL 60515

NRC Section Chief: Anthony J. Mendiola

Carolina Power & Light Company, et al., Docket No. 50–325, Brunswick Steam Electric Plant, Unit 1, Brunswick County, North Carolina

Date of amendment request: September 18, 2001.

Description of amendment request: The proposed amendments would change the Technical Specifications (TS) to revise the Minimum Critical Power Ratio (MCPR) Safety Limit values contained in TS 2.1.1.2, and revise the MCPR Safety Limit values from 1.10 to 1.12 for two recirculation loop operation and from 1.11 to 1.14 for single recirculation loop operation.

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

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

The proposed license amendment will establish MCPR Safety Limit values of 1.12 for two recirculation loop operation and 1.14 for single recirculation loop operation. The revised MCPR Safety Limit values have been determined using NRC-approved methods and procedures. These procedures incorporate cycle-specific parameters and reduced power distribution uncertainties in the determination of the MCPR Safety Limit values. These proposed MCPR Safety Limit values do not affect the operability of any plant systems nor do these revised values compromise any fuel performance limits. Therefore, the proposed change to the MCPR Safety Limit values does not result in an increase in the probability of a previously evaluated accident.

The consequences of a previously evaluated accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the accident, the availability and successful functioning of the equipment assumed to operate in response to the accident, and the setpoints at which these actions are initiated. The MCPR Safety Limit values are determined to ensure that 99.9 percent of the fuel rods will not experience boiling transition during any plant operation if the limit is not exceeded. Operational MCPR limits will be applied that ensure the MCPR Safety Limit is not exceeded during all modes of operation and anticipated operational occurrences. The MCPR Safety Limit does not impact the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated, and there are no adverse effects on the factors contributing to offsite and onsite dose. The proposed change to the MCPR Safety Limit values does not affect the performance of any equipment used to mitigate the consequences of a previously evaluated accident. Also, the proposed change does not affect setpoints that initiate protective or mitigative actions. Based on the determination of the MCPR Safety Limit values using conservative NRC-approved methods and the operability of plant systems designed to mitigate the consequences of accidents not being changed, the proposed changes to the MCPR Safety Limit values does not significantly increase the consequences of a previously evaluated accident.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This proposed license amendment does not involve any facility modifications, and plant equipment will not be operated in a different manner. Also, no new initiating events or transients result from the MCPR Safety Limit changes. As a result, no new failure modes are being introduced. Therefore, the proposed changes to the MCPR Safety Limit values will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety is established through the design of the plant structures, systems, and components; through the parameters within which the plant is operated; through the establishment of setpoints for actuation of equipment relied upon to respond to an event; and through margins contained within the safety analyses. The proposed change to the MCPR Safety Limit values does not adversely impact the performance of plant structures, systems, components, and setpoints relied upon to respond to mitigate an accident. The MCPR Safety Limit values have been calculated using NRC-approved methods and procedures. The MCPR Safety Limit values are determined to ensure that 99.9 percent of the fuel rods will not experience boiling transition during any plant operation if the limits are not exceeded, thereby ensuring that fuel cladding integrity is maintained. Based on the assurance that the fuel design criteria are being met, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

Dominion Nuclear Connecticut, Inc., Docket Nos. 50–245, 50–336, and 50–423, Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, New London County, Connecticut

Date of amendment request: August 8, 2001.

Description of amendment request: The proposed amendment would incorporate two changes into each operating license: (1) Revise the physical protection (security) related license condition to indicate that the physical security program plans listed, may, rather than do contain, safeguards information, and (2) change the name of the ''Millstone Nuclear Power Station'' to the ''Millstone Power Station.''

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

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

The first proposed clarification modifies the physical protection (security) related license condition within the respective operating license (OL) to indicate the physical security program plans listed, may, rather than do contain, safeguards information. The second proposed change to reflect the change in name of the facility from the ''Millstone Nuclear Power Station'' to the ''Millstone Power Station'' is editorial. Neither of these changes alter any regulatory requirements or have an impact on the acceptance criteria for any design basis accident described in the respective Unit 2 or 3 Updated Final Safety Analysis Report (UFSAR) or the Unit I Defueled Safety Analysis Report (DSAR).

These changes have no impact on plant equipment operation. Since the changes are solely an administrative or editorial change to the OL, they cannot affect the likelihood or consequences of accidents. Therefore, these changes will not increase the probability or consequences of an accident previously evaluated.

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

The proposed changes have no impact on plant operation. Since the proposed changes are solely an administrative or editorial change to the OL, they do not affect plant operation in any way.

The changes do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since the proposed changes are solely a clarification or an editorial change to the OL, they do not affect plant operation in any way. The proposed changes do not impact any acceptance criteria for the design basis accidents described in the respective Unit No. 2 or No. 3 UFSAR or the Unit No. 1 DSAR and do not impact the consequences of accidents previously evaluated. Therefore, the proposed changes will not result in a 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: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, Connecticut 06835.

NRC Section Chief: Stephen Dembek

Dominion Nuclear Connecticut Inc., et al., Docket Nos. 50–336 and 50–423, Millstone Nuclear Power Station, Unit Nos. 2 and 3, New London County, Connecticut

Date of amendment request: August 9, 2001.

Description of amendment request: The proposed amendments modify the Millstone Nuclear Power Station, Unit Nos. 2 (MP2) and 3 (MP3) Technical Specifications (TSs) to avoid confusion between the qualification standards of the facility staff, who are qualified to American National Standards Institute (ANSI) N18.1–1971/Regulatory Guide (RG) 1.8 Revision 0, and the operators who will be qualified to the education and experience guidelines outlined by National Academy for Nuclear Training ACAD 00–003 ''Guidelines for Initial Training and Qualification of Licensed Operators.''

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

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

The proposed administrative clarification modifies the Unit Nos. 2 and 3 TS to avoid confusion between the qualification standards of the facility staff, who are qualified to American National Standards Institute (ANSI), ''Selection and Training of Nuclear Power Plant Personnel,'' ANSI N18.1–1971/Regulatory Guide 1.8, Revision 0, ''Qualification and Training of Personnel for Nuclear Power Plants,'' and the operators who will be qualified to the education and experience guidelines outlined by National Academy for Nuclear Training (NANT 2000 Guidelines), ACAD 00–003, ''Guidelines for Initial Training and Qualification of Licensed Operators.'' The training of the operators themselves is not affected, this change only modifies the education and experience requirements they must meet to qualify for the operator training program. The reactor operator and senior reactor operator applicant (or upgrade) still must learn and are tested on the same material, demonstrate their proficiency on the facility simulator and meet other requirements. Consequently, this change has no impact on the capability of licensed operators, it only modifies and provides alternative qualifications for entry into the program.

This change will not alter any regulatory requirements or have an impact on the acceptance criteria for any design basis accident described in the respective Unit Nos. 2 or 3 Updated Final Safety Analysis Report (UFSAR).

The change has no impact on plant equipment operation. Since the change is solely an administrative change to the Technical Specifications, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase 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 change has no impact on plant operation. Since the proposed change is solely an administrative change to the Technical Specifications, it does not affect plant operation in any way.

The change does not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The change does not alter the way any structure, system, or component functions and does not alter the manner in which the plant is operated. The change does not introduce any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Since the proposed change is solely an administrative change to the Technical Specifications, it does not affect plant operation in any way.

The proposed change does not impact any acceptance criteria for the design basis accidents described in the respective Unit Nos. 2 or 3 UFSAR and does not impact the consequences of accidents previously evaluated. Therefore, the proposed change will not result in a 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 28, 2001.

Description of amendment request: The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 3 (MP3) Technical Specifications to remove the surveillance requirement associated with post maintenance testing of the containment isolation valves.

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

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

The proposed Technical Specification change to remove the surveillance requirement to perform post maintenance testing of the containment isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The containment isolation valves are not accident initiators. The proposed change will not revise the operability requirements (*e.g.*, valve stroke time) for the containment isolation valves. Proper operation of the containment isolation valves will still be verified, as appropriate, following maintenance activities. As a result, the design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 Final Safety Analysis Report, and the consequences of the design basis accidents will remain the same. Therefore, the proposed change will not increase 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 change to the Technical Specifications does not impact any system or component that could cause an accident. The proposed change will not alter the plant configuration (no new or different type of equipment will be installed) 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. The response of the plant and the operators following an accident will not be different. In addition, the proposed change does not introduce any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The proposed Technical Specification change to remove the surveillance requirement to perform post maintenance testing of the containment isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The operability requirements for the containment isolation valves have not been changed, and proper operation of the containment isolation valves will still be verified, as appropriate, following maintenance activities. The containment isolation valves will continue to be able to mitigate the design basis accidents as assumed in the safety analysis. Therefore, the proposed change will not result in a 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Waterford, CT 06141–5127.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 7, 2001.

Description of amendment request: The proposed amendment would change the Technical Specifications Regarding the Post Accident Monitoring Instrumentation (Table 3.3.3–1 of Section 3.3.3. ''Post Accident Monitoring Instrumentation''). Specifically, the proposed amendment would reword the number of required channels stated for the core exit thermocouples (CETs) to be the same as the Standard Technical Specifications; delete notes that describe the redundant channels for the Reactor Coolant System (RCS) Hot Leg Temperature, the RCS Cold Leg Temperature and Main Steam Line Radiation and modify the note pertaining to the redundant channel for Steam Generator Level (Wide Range) to clarify what Condition Statements apply when the instrument channel and/or the Auxiliary Feedwater Flow instrument channel is inoperable. Other existing notes in the Table are proposed to be renumbered to accommodate the above 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:

Consistent with the criteria of 10 CFR 50.92, the enclosed [proposed] application is judged to involve no significant hazards based on the following information:

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

Response: The proposed amendment involves rewording or clarification of technical specification requirements to properly reflect the design of post accident monitoring instrumentation at Indian Point 3. The proposed rewording of the required channels for core exit thermocouples adopts the wording from the Standard Technical Specifications, which is applicable to the Indian Point 3 design. The proposed deletion of Notes (a), (b), and (g) removes design information that is not needed for the specification to limit plant operation in response to inoperable instrument channels. The proposed rewording of Note (f) clarifies the existing requirement by making a more explicit statement about the applicable conditions for the affected functions.

Renumbering other Table notes is an editorial change to keep the notes in sequential order.

The proposed amendment does not involve any changes to plant equipment, setpoints, or the way in which the plant is operated. These changes do not affect accident initiators or accident mitigating systems. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: The proposed amendment involves rewording or clarification of technical specification requirements to properly reflect the design of post accident monitoring instrumentation at Indian Point 3. The proposed amendment does not involve any changes to plant equipment, setpoints, or the way in which the plant is operated. These changes do not affect accident initiators or accident mitigating systems. Therefore the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: The proposed amendment involves rewording or clarification of technical specification requirements to properly reflect the design and licensing basis of post accident monitoring instrumentation at Indian Point 3. The proposed rewording of the required channels for core exit thermocouples adopts the wording from the Standard Technical Specifications, which is applicable to Indian Point 3. This will ensure that appropriate condition statements are entered in the event that core exit thermocouples become inoperable. Notes (a), (b), and (g) provide design information that is not needed in the specification for plant operators to enter appropriate condition statements when inoperable instrument channels in the affected functions are identified. The rewording of Note (f) more clearly states the existing requirement and makes no change to the required actions or completion times for the associated inoperable instrument channels.

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

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

Attorney for licensee: Mr. John Fulton, Assistant General Counsel, Entergy Nuclear Generating Station, 600 Rocky Hill Road, Plymouth, MA 02360.

NRC Section Chief: Lakshminaras Raghavan, Acting.

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: August 13, 2001.

Description of amendment request: The proposed amendments would revise technical specifications (TS) to support a planned upgrade to the reactor water level instrumentation. Currently, many low-level actuation functions use Yarway level indicating switches. This includes emergency core cooling system (ECCS), reactor core isolation cooling (RCIC) and feedwater systems. The Yarways will be replaced with more reliable analog level transmitters and additional electronic trip units. The upgrade will provide sensing devices for reactor vessel water level signals and indications that are more reliable with less drift and will require less frequent surveillance requirements. The proposed changes align the TS surveillance requirements with the instrumentation upgrades. This includes changes to calibration frequencies, functional testing and allowable values.

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

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

During the upcoming refueling outages at Quad Cities Nuclear Power Station (Unit 1 and Unit 2), a design change will be implemented that upgrades the existing reactor vessel level trip instrumentation used in various applications at Quad Cities Nuclear Power Station, including the Emergency Core Cooling System (ECCS), Reactor Core Isolation Cooling System (RCIC) and Feedwater systems.

Technical Specification (TS) requirements that govern operability or routine testing of plant instruments are not assumed to be initiators of any analyzed event because these instruments are intended to prevent, detect, or mitigate accidents. Therefore, these changes will not involve an increase in the probability of occurrence of an accident previously evaluated. Additionally, these changes will not increase the consequences of an accident previously evaluated because the proposed change does not adversely impact structures, systems, or components (SSCs). The planned instrument upgrade is a more reliable design than existing equipment. The proposed TS change maintains existing requirements that ensure components are operable when necessary for the prevention or mitigation of accidents or transients. Revised allowable values for the associated functions have been established in accordance with EGC's setpoint methodology, which is consistent with industry standards. The setpoint methodology establishes TS allowable values that assure systems structures and components (including initiation and trip functions) respond in a manner consistent with the plant safety analysis. Furthermore, there will be no change in the types or significant increase in the amounts of any effluents released offsite. For these reasons, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed changes support a planned instrumentation upgrade. The change provides revised Surveillance Requirements to ensure operability. The change does not adversely impact the manner in which the instrument will operate under normal and abnormal operating conditions. These changes reflect the improved performance of the instrumentation upgrade and provide an equivalent level of safety. The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change supports a planned instrumentation upgrade. The proposed change does not affect the probability of failure or availability of the affected instrumentation. The change to an analog trip system to monitor reactor vessel level provides for increased reliability. The change has no impact on the underlying design functions. The proposed TS surveillance requirements are consistent with current TS requirements for functions that employ analog trip unit devices. The proposed allowable values have been established in accordance with EGC's setpoint methodology, which considers instrument design and performance characteristics. The methodology establishes TS allowable values with sufficient margin to assure that the plant safety analysis assumptions (e.g., certain initiation and trip functions) are maintained. As such, the trip and actuation functions continue to ensure design basis requirements are maintained. Therefore, it is concluded that the proposed changes will not result in a 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.

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

Date of amendment request: June 29, 2001.

Description of amendment request: The proposed license amendment would change the technical specifications (TSs) to reflect revised reactor coolant system (RCS) heatup and cooldown pressure and temperature (P/T) limit curves that will be valid through 22 effective full power years (EFPYs). The overpressure protection system (OPPS) power-operated relief valve (PORV) setpoints and the OPPS enabling temperature would also be revised. The proposed BVPS–1 P/T limits incorporate the results from the testing of the Capsule Y described in WCAP–15571, ''Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program,'' Revision 0, November 2000. These changes have been prepared using the Nuclear Regulatory Commission-approved methodology described in WCAP–14040–NP–A, ''Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'' Revision 2, January 1996, with two exceptions. These exceptions include the use of (1) the American Society of Mechanical Engineers (ASME) Code Case N–640, ''Alternate Reference Fracture Toughness for Development of P–T Curves for Section XI, Division 1,'' March 1999, and (2) the ASME Boiler and Pressure Vessel Code, Section XI, ''Rule for Inservice Inspection of Nuclear Power Plant Components,'' Appendix G, ''Fracture Toughness Criteria for Protection Against Failure,'' December 1995 (through 1996 Addendum). The TS Bases and Figure Index will also be changed to reflect 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. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No. The proposed changes do not result in physical changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. Changing the heatup and cooldown curves, power operated relief valve (PORV) setpoint and overpressure protection system (OPPS) enable temperature to reflect 22 effective full power years (EFPY) will not affect the ability of the OPPS to control the reactor coolant system (RCS) at low temperatures such that the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure/temperature (P/T) limits. These changes were determined in accordance with the methodologies set forth in the regulations to provide an adequate margin of safety to ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events.

Also, the proposed changes do not impact the design of plant systems such that previously analyzed SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

The proposed changes do not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the UFSAR. As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

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

No. The proposed changes do not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the technical specification. The proposed changes do not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor do they alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

The proposed changes do not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed changes have no effect on previously evaluated accidents.

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

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

No. The proposed changes have been determined through supporting analyses to be in accordance with the methodologies set forth in the regulations. Compliance with NRC approved methodologies provide for an adequate margin of safety and ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events as described in the UFSAR.

The new heatup and cooldown curves define the limits for ensuring prevention of nonductile failure for the BVPS Unit No. 1 reactor vessel and do not significantly reduce the margin of safety for the plant.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: L. Raghavan (Acting).

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

Date of amendment request: September 27, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to (1) change the diesel fuel supply volume required for diesel generator (DG) operability, (2) clarify existing wording, (3) add a TS limiting condition for operation (LCO) and a TS surveillance requirement (SR) regarding DG air receivers, (4) delete a current TS SR concerning DG starting air compressors, and (5) restructure and renumber the TS LCOs and SRs for applicability and administrative purposes.

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

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

The proposed Technical Specification changes do not introduce new equipment or new equipment operating modes, nor do the proposed changes alter existing system relationships. The proposed amendment does not introduce new failure modes.

The proposed revision to the Monticello TS[s] renumbers and relocates TS[s] as appropriate to provide a more understandable TS, deletes an existing TS SR which does not satisfy the requirements of 10 CFR 50.36 for inclusion in the TS[s], adds a new TS LCO and SR for DG air start receivers which more appropriately complies with the requirements of 10 CFR 50.36, and revises the minimum number of gallons of diesel fuel required in the Diesel Oil Storage Tank for the DG to be declared operable.

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

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

The proposed changes do not introduce a new mode of plant operation, or involve a physical modification to the plant. The proposed Technical Specification changes do not introduce new equipment, nor do the proposed changes alter existing system relationships. The proposed amendment does not introduce new failure modes.

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

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

The proposed changes maintain the current TS requirements for safe operation of the Monticello plant. The proposed changes do not involve a physical modification to the plant, or a new mode of operation. The proposed changes do not alter the scope of equipment currently required to be operable nor do the proposed changes affect equipment safety functions. The proposed Technical Specification changes do not introduce new equipment, nor do the proposed changes alter existing system relationships. The proposed amendment does not introduce new failure modes.

Therefore, these proposed changes will not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: William D. Reckley

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: August 17, 2001.

Description of amendment request: The proposed Technical Specifications (TS) change will: (1) modify Salem TS surveillance requirement 4.6.2.3, and (2) revise the associated TS Bases. Specifically, the proposed change will modify the current acceptance criterion for the service water flow rate through the Containment Fan Coil Units from ≥ 2,550 gallons per minute (gpm) to ≥ 2,300 gpm.

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

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

The containment ventilation system, including the containment fan coil units is not an accident initiator.

The proposed TS change to modify the Salem TS surveillance requirement 4.6.2.3 to the service water-cooling water flow through the fan coil units is bounded by the present licensing and design bases analyses. The new proposed flow rate, in conjunction with its associated heat exchanger thermal fouling factor, will continue to maintain the assumed minimum containment heat removal capability to be within the Salem Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses. Therefore, the proposed change will not involve a significant increase in 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 TS change to modify the Salem TS surveillance requirement 4.6.2.3 to the service water-cooling water flow though the fan coil units is bounded by the present licensing and design bases analyses. The manner and frequency at which the surveillance test is conducted remains unchanged. The physical facility remains unchanged.

Therefore, the new proposed flow rate does not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The proposed TS change to modify the Salem TS surveillance requirement 4.6.2.3 to the service water-cooling water flow though the fan coil units is bounded by the present licensing and design bases analyses. The new proposed flow rate, in conjunction with its associated heat exchanger thermal fouling factor, will continue to maintain the assumed minimum containment heat removal capability to be within the Salem UFSAR Chapter 15 analyses. Consequently, the existing margins of safety with respect to the current design-basis assumptions of pressure of 47 psig and a saturation temperature of 271 °F in containment during a design-basis accident is maintained.

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 consideration.

Attorney for licensee: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–321, Edwin I. Hatch Nuclear Plant, Unit 1, Appling County, Georgia

Date of amendment request: August 31, 2001.

Description of amendment request: The proposed amendment would allow a one-time deferral of the Type A Containment Integrated Leak Rate Test based on the risk-informed guidance in Regulatory Guide 1.174.

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 Technical Specification change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to Technical Specification 5.5.12 (''Primary Containment Leakage Rate Testing Program'') involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such the reactor containment itself and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications. Industry experience has shown, as documented in NUREG–1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. HNP Unit 1 ILRT test history supports this conclusion. NUREG–1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. The integrity of the reactor containment is subject to two types of failure mechanisms which can be categorized as (1) activity based and (2) time based. Activity based failure mechanisms are defined as degradation due to system and/or component modifications or maintenance. Local leak rate test requirements and administrative controls such as design change control and procedural requirements for system restoration ensure that containment integrity is not degraded by plant modifications or maintenance activities. The design and construction requirements of the reactor containment itself combined with the containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the containment coatings program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing requirements invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific requirements and conditions of the Primary Containment Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications.

HNP Unit 1 and industry experience strongly supports the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with ASME Section XI, the Maintenance Rule and the Coatings Program serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by Type A testing. Additionally, the on-line containment monitoring capability that is inherent to inerted BWR containments allows for detection of gross containment leakage that may develop during power operation. The combination of these factors ensures that the margin of safety that is inherent in plant safety analysis is maintained. Therefore, the proposed Technical Specification change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: August 31, 2001.

Description of amendment request: The proposed amendments would extend the completion times for the required actions associated with restoring an inoperable emergency diesel generator.

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 extend the Technical Specifications required Completion Times for restoration of an inoperable emergency diesel generator (DG) to a maximum of 14 days. Additionally, the proposed extension of the Completion Time to 14 days results in a corresponding extension of the time period associated with discovery of failure to meet Limiting Condition for Operation 3.8.1 to 17 days. (This provides a maximum time limit for overlapping inoperabilities of DGs and offsite sources.)

For both Plant Hatch units A and C DGs, to utilize the 72 hours to 14 day period of the proposed extended Completion Time, compensatory action is required to ensure two DGs per unit remain available. This action consists of dedicating the 1B DG to that unit with the inoperable DG. This means that the 1B DG will be inhibited from an automatic swap to the opposite unit when that unit (the non-maintenance unit) experiences and undervoltage condition on its F 4160 volt bus, regardless of the presence or absence of a loss of coolant accident (LOCA) signal. Inhibiting the automatic transfer makes the 1B DG inoperable (with a Completion Time of 14 days) for the non-maintenance unit.

Completion Times are not an initial condition or assumption of any analyzed event. DGs are not initiators of any analyzed event. No event mitigation assumes more than two DGs per unit. The consequences of an accident are independent of the time the DGs are out of service provided adequate DG availability is assured. Compensatory actions are proposed in this amendment request that ensure adequate DG availability for both Plant Hatch units. Therefore, the assumptions regarding DG available are maintained.

To fully evaluate the effect of the proposed DG Completion Time extension, Probabilistic Safety Assessment methods and a deterministic analysis were utilized. The results of the analyses show no significant increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF).

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

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

The proposed changes do involve a change to the plant configuration when either unit's A or C DG is utilizing the extended Completion Time (*i.e.,* inoperable in excess of 72 hours). That configuration change ensures that both units have two dedicated DGs. Furthermore, affixing the 1B DG to one unit will cause it (1B DG) to be inoperable with respect to the Technical Specifications. Ensuring two DGs available to each unit for event mitigation in no way creates the possibility of a new or different type of accident.

No other change in the design, configuration, or method of operation of the plant is introduced by the proposed change. The changes do not alter any assumptions made in the safety analyses. No new failure modes are introduced.

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

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

Since all assumptions of the plant event analyses are maintained, there is no effect on the margin of safety in any safety analyses. If there is any margin of safety ascribed to DG availability and plant risk, it has been determined that such a margin of safety is not significantly reduced, as the proposed changes have been evaluated both deterministically and using a risk-informed approach. These evaluations concluded the following with respect to the proposed changes:

Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, sufficient safety margins will be maintained, and any increases in CDF and LERF are small and consistent with the NRC Safety Goal Policy Statement (**Federal Register**, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986, as interpreted by NRC Regulatory Guides 1.174 and 1.177). Furthermore, increases in risk posed by potential combinations of equipment out of service during the proposed DG extended Completion Time will be managed by the site configuration risk management procedure, consistent with 10 CFR 50.65, ''Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,'' paragraph (a)(4).

The availability of offsite power together with the availability of the other DGs and the use of on-line risk assessment tools provide adequate compensation for the potential small incremental increase in plant risk of the extended DG Completion Time. In addition, the increased availability of the DGs during refueling outages offsets the small increase in plant risk during operation. The proposed extended DG Completion Times, in conjunction with the availability of the other DGs continues to provide adequate assurance of the capability to provide power to the engineered safety features buses. Therefore, implementation of the proposed changes will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Section Chief: Richard J. Laufer, Acting.

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

Date of amendment request: September 7, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification Section 3.6.11, ''Ice Bed,'' Surveillance Requirement (SR) 3.6.11.2, SR 3.6.11.3, and the associated Bases, to lower the minimum required average ice basket weight from 1236 pounds to 1110 pounds, and the corresponding total weight of the stored ice in the ice condenser from 2,403,800 pounds to 2,158,000 pounds.

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 primary purpose of the ice bed is to provide a large heat sink to limit peak containment pressure in the event of a release of energy from a design basis loss-of-coolant (LOCA) or high energy line break (HELB) in containment. The LOCA requires the greatest amount of ice compared to other accident scenarios, therefore the reduction in ice weight is based on the LOCA analysis. The amount of ice in the bed has no impact on the initiation of an accident, but rather on the mitigation of the accident.

The containment integrity analysis shows that the proposed reduced ice weight is sufficient to maintain the peak containment pressure below the containment design pressure, and that the containment heat removal systems function to rapidly reduce the containment pressure and temperature in the event of a LOCA. Therefore, containment integrity is maintained and the consequences of an accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR) are not significantly increased.

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

The ice condenser serves to limit the peak pressure inside containment following a LOCA. TVA has evaluated the revised containment pressure analysis and determined that sufficient ice would be present to maintain the peak containment pressure below the containment design pressure. Therefore, the reduced ice weight does not create the possibility of an accident that is different than any already evaluated in the WBN UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed change.

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

The containment integrity analysis for reduced ice weight results in a peak containment pressure that is slightly lower than that in the previous analysis of record. This reduction in peak pressure, along with the ice weight reduction, is due to the removal of analytical conservatism combined with a better segmental representation of the mass and energy release transient from the computer models.

The revised technical specifications ice weight surveillance limits are based on the ice weight assumed in the containment integrity analysis, with margin included for sublimation that is based on actual sublimation data from the first three refueling cycles at WBN. The analysis further demonstrates that the existing relationship between ice bed melt-out and containment spray switchover has been conservatively maintained. With the reduced ice inventory, melt-out of the ice bed following a worst case large break LOCA has been determined to occur after the switchover of containment spray to the recirculation mode. Thus, the reduced ice bed mass does not result in a reduction in the margin for operator action to effect the switchover.

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.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Units 1 and 2, Somervell County, Texas

Date of amendment request: August 24, 2001.

Brief description of amendments: The proposed change would revise Comanche Peak Steam Electric Station, Units 1 and 2, Technical Specification (TS) 3.3.2, entitled ''ESFAS [Engineered Safety Features Actuation System] Instrumentation,'' and TS 3.3.6, entitled ''Containment Ventilation Isolation Instrumentation,'' to change the surveillance frequency for Westinghouse Electric Company-type AR relays, used as Solid State Protection System slave relays or auxiliary relays, from quarterly to refueling outage frequency. Surveillance Requirements 3.3.2.6 and 3.3.6.5 would be revised to change the frequency from ''92 days'' to ''92 days *OR* 18 months for Westinghouse type AR relays.''

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

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

Response: No.

The proposed change to the Technical Specifications does not result in a condition where the design, material, or construction standards that were applicable prior to the change are altered. The same ESFAS instrumentation is being used and the same ESFAS system reliability is expected. The proposed change will not modify any system interface or function and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent the performance of any accident mitigation systems or alter any assumptions previously made in evaluating the radiological consequences of an accident as described in the safety analysis report.

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

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

Response: No.

The proposed change does not alter the performance of the ESFAS mitigation systems assumed in the plant safety analysis. Changing the interval for periodically verifying ESFAS slave relays (assuring equipment operability) will not create any new accident initiators or scenarios.

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

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

Response: No.

The proposed change does not affect the total ESFAS system response assumed in the safety analysis. The periodic slave relay functional verification is relaxed because of the demonstrated high reliability of the relay and its insensitivity to any short term wear or aging effects.

Therefore the proposed change does not involve a 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: George L. Edgar, Esq., Morgan, Lewis and Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 27, 2001 (WO 01–0038)

Description of amendment request: The proposed amendment would revise Section 5.3.1.1 of the Technical Specifications to replace the current qualifications in ANSI/ANS 3.1–1981 for licensed operators and senior operators with the National Academy for Nuclear Training, ''Guidelines for Initial Training and Qualification of Licensed Operators,'' dated January 2000.

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

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

The proposed TS change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program. [The change conforms] to the current requirements of 10 CFR 55.

Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. WCNOC's [Wolf Creek Nuclear Operating Corporation's] licensed operator training program is accredited by INPO [Institute for Nuclear Power Operations] and is based on a system[']s approach to training. The proposed TS change takes credit for the INPO accreditation of the licensed operator training program. The TS requirements for all other unit staff qualifications remain unchanged.

Therefore, the proposed change does not involve a signification 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 proposed TS change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program and to conform to the revised 10 CFR 55.

As noted above, although licensed operator qualifications and training may have an indirect impact on the possibility of a new or different kind of accident from any accident previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised [10 CFR 55] rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and based on a system[']s approach to training. As previously noted, WCNOC's licensed operator training program is accredited by INPO and is based on a system[']s approach to training. The proposed TS change takes credit for the INPO accreditation of the licensed operator training program. The TS requirements for all other unit staff qualifications remain unchanged.

Additionally, the proposed TS change does not affect plant design, hardware, system operation, or procedures. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

The proposed TS change is an administrative change to clarify the current requirements applicable to licensed operator qualifications and licensed operator training program. This change is consistent with the requirements of 10 CFR 55. The TS qualification requirements for all other unit staff remain unchanged.

Licensed operator qualifications and training can have an indirect impact on a margin of safety. However, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR 55 [rule], determined that this impact remains acceptable when licensees maintain a licensed operator training program that is accredited and based on a system[']s approach to training. As noted previously, WCNOC's licensed operator training program is accredited by INPO and is based on a system[']s approach to training.

The NRC has concluded, as stated in NUREG–1262, ''Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses,'' that the standards and guidelines applied by INPO in their training accreditation program are equivalent to those put forth or endorsed by the NRC. As a result, maintaining an INPO accredited, systems approach based licensed operator training program is equivalent to maintaining an NRC approved licensed operator training program which conform with applicable NRC Regulatory Guides or

NRC endorsed industry standards. The margin of safety is maintained by virtue of maintaining an INPO accredited licensed operator training program.

In addition, the NRC has recently published NRC Regulatory Issue Summary 2001–01, ''Eligibility of Operator License Applicants,'' dated January 18, 2001, ''to familiarize addresses with the NRC's current guidelines for the qualification and training of reactor operator (RO) and senior operator (SO) license applicants.'' This document again acknowledges that the INPO National Academy for Nuclear Training (NANT) guidelines for education and experience, outline acceptable methods for implementing the NRC's regulations in this area.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: August 14, 2001, as supplemented on August 21, 2001.

Description of amendment request: The proposed amendment would extend the allowed outage time for the high pressure coolant injection (HPCI) and reactor core isolation cooling systems from 7 days to 14 days. Requirements were added to immediately assure the availability of alternate means of high pressure coolant makeup. Also clarifying changes were made to Technical Specification (TS) 3.5.E.2 and TS 3.5.G.2 by reformatting the TSs to make nomenclature consistent regarding HPCI and the automatic depressurization system (ADS) as being systems not subsystems.

Date of publication of individual notice in **Federal Register:** September 16, 2001 (66 FR 48152).

Expiration date of individual notice: October 18, 2001.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: January 29, 2001, as supplemented July 6, 2001.

Brief description of amendment: The amendment removes the note from TMI–1 Technical Specification 4.5.4.1 which restricts the applicability of the specified Engineered Safeguards Feature (ESF) Systems leakage rate limit of 15 gallons per hour to the current operating cycle (Cycle 13). The amendment also approves full scope implementation of an alternate source term for TMI–1 in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.67.

Date of issuance: September 19, 2001.

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

Amendment No.: 235.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31703).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 19, 2001.

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: January 23, 2001, as supplemented August 22 and September 17, 2001.

Brief description of amendment: The amendment revised the TMI–1 Technical Specification requirements for containment integrity associated with the personnel and emergency air locks during fuel movement and refueling operations. Partial implementation of an alternate source term (AST) in accordance with Regulatory Guide 1.183, ''Alternate Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,'' and Title 10 of the Code of Federal Regulations, Section 50.67, which the licensee had also requested in its application, was not necessary because the Commission approved full implementation of an AST for TMI–1 in Amendment No. 235 dated September 19, 2001.

Date of issuance: October 2, 2001.

Effective date: As of its date of issuance, contingent upon the licensee's implementation of regulatory commitments contained in the licensee's letters dated August 22 and September 17, 2001, and shall be implemented within 30 days of issuance.

Amendment No.: 236.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31702).

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

No significant hazards consideration comments received: No.

AmerGen Energy Company, LLC, Docket No. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania

Date of application for amendment: December 6, 2000, as supplemented July 13 and September 6, 2001.

Brief description of amendment: The amendment revises the once-through steam generator (OTSG) surveillance criteria contained in the TMI–1 Technical Specifications (TSs) to allow OTSG tubes to remain in service with indications of inside diameter intergranular attack located below the upper tubesheet secondary face. The changes also extend the repair criteria from a cycle-to-cycle basis to a permanent basis.

Date of issuance: October 5, 2001.

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

Amendment No.: 237.

Facility Operating License No. DPR–50: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 24, 2001 (66 FR 7669).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 15, 2001.

Brief description of amendments: The amendments delete Technical Specifications Section 5.5.3, ''Post Accident Sampling System,'' for Palo Verde Nuclear Generating Station, Units Nos. 1, 2 and 3, and thereby eliminate the requirements to have and maintain the post-accident sampling system.

Date of issuance: September 28, 2001.

Effective date: September 28, 2001, and shall be implemented within 7 months of the date of issuance.

Amendment Nos.: Unit 1–136, Unit 2–136, Unit 3–136.

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

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41611).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 1, 2001, as supplemented August 20, 2001.

Brief description of amendments: The amendments change the Technical Specifications related to the pressure-temperature limit curves.

Date of issuance: October 4, 2001.

Effective date: October 4, 2001.

Amendment Nos.: 214 and 241.

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

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29350). The August 20, 2001, supplement contained clarifying information only, and did not change the initial no significant hazards consideration determination, or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 17, 2001, as supplemented March 23 and August 31, 2001.

Brief description of amendments: The amendments change the Technical Specifications to relax the 24-month surveillance frequency of excess flow check valves (EFCVs) by limiting the number of tests to a representative sample every 24 months such that each EFCV will be tested at least once every 10 years.

Date of issuance: October 4, 2001.

Effective date: October 4, 2001.

Amendment Nos.: 215 and 242.

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

Date of initial notice in **Federal Register:** February 21, 2001 (66 FR 11052). The March 23 and August 31, 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 amendments is contained in a Safety Evaluation dated October 4, 2001.

No significant hazards consideration comments received: No.

Carolina Power & Light Company, Docket No. 50–400, Shearon Harris Nuclear Power Plant (HNP), Wake and Chatham Counties, North Carolina

Date of application for amendment: December 13, 2000, as supplemented on February 9, and August 3, 2001.

Brief description of amendment: This amendment revises Technical Specification (TS) 3/4.9.2, ''Refueling Operations—Instrumentation,'' and the associated Bases to permit using one Source Range Nuclear Flux Monitor and one Wide Range Neutron Flux Monitor during MODE 6 (Refueling) instead of the two Source Range Nuclear Flux Monitors specified in the current HNP TS.

Date of issuance: September 10, 2001.

Effective date: September 10, 2001.

Amendment No. 105.

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

Date of initial notice in **Federal Register:** January 24, 2001 (66 FR 7672).

The February 9, and August 3, 2001, submittals 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 September 10, 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: December 14, 2000, as supplemented August 16, and September 12, 2001.

Brief description of amendment: The amendment revises Technical Specification 3/4.8.1 related to Emergency Diesel Generators (EDGs), and specifically revises Surveillance Requirement 4.8.1.1.2.f.7, the 24-hour EDG endurance run test, by removing the restriction to perform the test during shutdown conditions.

Date of issuance: October 3, 2001.

Effective date: October 3, 2001.

Amendment No.: 106.

Facility Operating License No. NPF–63: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 24, 2001 (66 FR 7673). The August 16, and September 12, 2001 supplements contained clarifying information that did not change the scope of the December 14, 2000, application nor the proposed initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: October 26, 2000, as supplemented by letters dated February 9, February 28, March 14, March 15, March 23, May 2, July 13, July 17, and August 2, 2001.

Brief description of amendment: The amendment revises the Defueled Technical Specifications to reflect the removal of the original 75-ton Reactor Building gantry crane and its replacement with an upgraded single-failure proof crane.

Date of issuance: September 28, 2001.

Effective date: The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment No.: 122.

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

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22025). The supplemental letters dated May 2, July 13, July 17, and August 2, 2001, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards considerations comments received: No.

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

Date of application for amendment: December 29, 2000, as supplemented May 2 and July 19, 2001.

Brief description of amendment: The amendment revises the Fermi 2 Technical Specifications associated with handling irradiated fuel assemblies, based on reevaluation of the design-basis fuel handling accident analysis with an alternative radiological source term.

Date of issuance: September 28, 2001.

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

Amendment No.: 144.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** February 7, 2001 (66 FR 9381). The application was renoticed on August 27, 2001 (66 FR 45062), due to supplemental information beyond the scope of the initial notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards consideration comments received: No.

Duke Energy Corporation, et al., Docket No. 50–414, Catawba Nuclear Station, Unit 2, York County, South Carolina

Date of application for amendment: March 9, 2001, as supplemented by letters dated July 25, September 10, and September 13, 2001.

Brief description of amendment: The amendment revised the cold leg elbow tap flow coefficients used in the determination of Reactor Coolant System flow rate. There are no changes to the associated Technical Specifications with this amendment.

Date of issuance: October 2, 2001.

Effective date: As of its date of issuance and shall be implemented before the startup of Cycle 12, and will be in effect only for the duration of Cycle 12.

Amendment No.: 186.

Facility Operating License No. NPF–52: Amendment did not revise the Technical Specifications.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34281). The supplements dated July 25, September 10, and September 13, 2001, provided clarifying information that did not change the scope of the March 9, 2001, application nor the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Date of application of amendments: July 18, 2000, supplemented August 22 and November 8, 2000, and June 7, July 26, and September 5, 2001.

Brief description of amendments: The amendments revised the Technical Specifications to incorporate provisions of the Automatic Feedwater Isolation System.

Date of Issuance: September 26, 2001.

Effective date: As of the date of issuance. It shall be implemented for each Oconee unit prior to reactor startup following installation of the system and training of appropriate personnel.

Amendment Nos.: 320, 320, and 320.

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

Date of initial notice in **Federal Register:** September 20, 2000 (65 FR 56949). The supplements dated August 22 and November 8, 2000, and June 7, July 26, and September 5, 2001, provided clarifying information that did not change the scope of the July 18, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 26, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: January 24, 2001, as supplemented by letter dated September 24, 2001.

Brief description of amendment: The amendment revised the Technical Specifications (TSs) to incorporate the provisions to perform routine diesel generator (DG) monthly testing by gradually accelerating the DG to operating speed. In addition, a new TS was added to require fast starts of the DGs on a 184-day frequency.

Date of issuance: September 27, 2001.

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

Amendment No.: 121.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 7, 2001 (66 FR 13801). The supplemental letter dated September 27, 2001, provided additional information that did not expand the scope of the NRC staff's initial proposed no significant hazards consideration determination (66 FR 13801, published March 7, 2001).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: May 23, 2001, as supplemented by letters dated July 23, 2001, and August 23, 2001.

Brief description of amendment: The amendment changes the following Technical Specifications (TSs): (1) the value of the safety limit minimum critical power ratio was changed in TS 2.1.1.2, (2) an editorial clarification to TS 5.6.5.a.5) was added to include the applicable reactor protection system instrumentation function, and (3) the list of the approved methodologies in TS 5.6.5.b. and the associated Bases and References were updated.

Date of issuance: October 3, 2001.

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

Amendment No.: 122.

Facility Operating License No. NPF–47: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34281). The supplemental letters dated July 23, 2001, and August 23, 2001, provided additional information that did not expand the scope of the application or change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2001.

No significant hazards consideration comments received: No.

Entergy Nuclear Generation Company, Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of application for amendment: May 31, 2001.

Brief description of amendment: The amendment revised Technical Specification 5.5.6, ''Technical Specification (TS) Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 which were published in the **Federal Register** (64 FR 53582) on October 4, 1999.

Date of issuance: October 2, 2001.

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

Amendment No.: 192.

Facility Operating License No. DPR–35: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 25, 2001 (66 FR 38761). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 7, 2000, as supplemented December 29, 2000.

Brief description of amendment: The changes revise Technical Specification Section 3.7.B.4 to allow a one-time replacement of Station 125V DC batteries 31 and 32 while at power. The one-time change is necessary to support an on-line replacement of the existing batteries with new batteries. In addition, a change is made on a one-time basis to conduct testing the battery while the plant is not shutdown. Also included is an administrative change involving the deletion of an expired one-time limiting condition for operation statement related to an Emergency Diesel Generator Fuel Oil Storage Tank repair effort.

Date of issuance: September 19, 2001.

Effective date: September 19, 2001.

Amendment No.: 208.

Facility Operating License No. DPR–64: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 15, 2000 (66 FR 15922).

The December 29, 2000, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 31, 2001.

Brief description of amendment: The amendment revised and transferred the inservice testing portion of Technical Specification (TS) 4.0.5 to TS 6.5.8, and eliminated the inservice inspection portion of TS 4.0.5. In addition, other sections of the TSs that reference TS 4.0.5 were revised to be consistent with the revisions discussed above.

Date of issuance: September 24, 2001.

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

Amendment No.: 233.

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

Date of initial notice in **Federal Register:** August 22, 2001 (66 FR 44167).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 2001.

No significant hazards consideration comments received: No.

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

Date of amendment request: May 22, 2001.

Brief description of amendment: The amendment changes Technical Specifications (TS) 3/4.7.1.2, Emergency Feedwater System, and expands and clarifies the current TS.

Date of issuance: October 4, 2001.

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

Amendment No.: 173.

Facility Operating License No. NPF–38: The amendment revised the Technical Specifications and Surveillance Requirements.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34283).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 23, 2001.

Brief description of amendments: The amendments deleted the loose parts monitoring system from the LGS Units 1 and 2 Technical Specifications and Bases. The amendments were based on the conclusions of the Boiling Water Reactor Owners' Group TopicalReport NEDC–32975P, ''Regulatory Relaxation for BWR Loose Parts Monitoring System,'' which was approved by the Nuclear Regulatory Commission's Safety Evaluation dated January 25, 2001.

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

Effective date: September 19, 2001.

Amendment Nos.: 153 and 117.

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

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41619).

The Commission's related evaluation of the amendments is contained in a safety evaluation dated September 19, 2001.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, PSEG Nuclear LLC, and Atlantic City Electric Company, Docket No. 50–278, Peach Bottom Atomic Power Station, Unit 3, York County, Pennsylvania

Date of application for amendment: May 30, 2001 (two letters), as supplemented July 24 (two letters), and August 13, 2001.

Brief description of amendment: This amendment revises Technical Specification 5.5.12 to allow a one-time change in the containment integrated leak rate test interval from the current 10 years to a test interval of 15 years.

Date of issuance: October 4, 2001.

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

Amendment No.: 244.

Facility Operating License No. DPR–56: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 11, 2001 (66 FR 36341). The July 24 (two letters), and August 13, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 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: March 28, 2001, as supplemented by letters dated May 18, June 15, and July 18, 2001.

Brief description of amendment: The amendment approves changes to the BVPS–1 Technical Specification boron concentration limits for the refueling water storage tank, accumulators, boron injection tank (BIT), and the reactor coolant system/refueling canal during Mode 6. In conjunction with the reduction in the maximum boron concentration in the BIT, the temperature controls on the BIT are eliminated.

Date of issuance: September 24, 2001.

Effective date: Immediately and to be implemented within 60 days.

Amendment No: 242.

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

Date of initial notice in **Federal Register:** July 25, 2001 (66 FR 38763). The May 18, June 15, and July 18, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 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, Beaver County, Pennsylvania

Date of application for amendment: March 28, 2001.

Brief description of amendment: These amendments approved reductions in the reactor coolant system and secondary coolant system specific activity limits specified in TS 3/4.4.8, ''Reactor Coolant System Specific Activity,'' and TS 3/4.7.1.4, ''Plant Systems Activity.'' These TS changes support revised safety analyses of the design-basis main steam line break dose consequence analysis, which assumes higher primary-to-secondary accident induced leakage in accordance with the methodology described in Generic Letter 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes by Outside Diameter Stress Corrosion Cracking.'' These amendments also authorized Updated Final Safety Analysis Report (UFSAR) changes.

Date of issuance: September 28, 2001.

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

Amendment No: 244.

Facility Operating License No. DPR–66: Amendment revised the Technical Specifications and authorized changes to the UFSAR.

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29354).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 18, 2001, as supplemented by letters dated February 20, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 2001.

Brief description of amendment: A portion of this amendment approves revisions to BVPS–2 TS 3/4.4.9, ''Pressure/Temperature Limits,'' heatup and cooldown curves.

Date of issuance: September 24, 2001.

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

Amendment No.: 122.

Facility Operating License No. NPF–73: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 27, 2001 (66 FR 39211). The February 20, April 12, May 7, May 18, June 9 (3 letters), June 26, June 29, August 21, and September 5, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 24, 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: July 18, 2001, as supplemented August 30 and September 6, 2001.

Brief description of amendments: The amendments revised Technical Specification (TS) 3.9.4 and its associated Bases to allow the containment equipment door to be open during core alterations or movement of non-recently irradiated fuel within the containment, provided that the capability for closure is maintained.

Date of issuance: September 27, 2001.

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

Amendment Nos: 216 and 210.

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

Date of initial notice in **Federal Register:** August 8, 2001 (66 FR 41622). The August 30 and September 6, 2001, submittals provided clarifying information that did not change the scope of the July 18, 2001, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 27, 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: September 26, 2000, as supplemented February 1, 2001, June 29, 2001, and August 10, 2001.

Brief description of amendments: The amendments would approve changes to revise the current licensing basis, as stated in the updated final safety analysis report, to require operator action to mitigate the effects of a loss of seal injection cooling to the reactor coolant pumps.

Date of issuance: September 28, 2001.

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

Amendment Nos.: 255 and 238.

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

Date of initial notice in **Federal Register:** October 18, 2000 (65 FR 62386) The February 1, June 29, and August 10, 2001, supplemental letters did not change the scope of the proposed action and did not change the NRC's preliminary no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: April 30, 2001, as supplemented June 27 and August 3, 2001.

Brief description of amendment: The amendment conforms the license to reflect the transfer of Facility Operating License No. DPR–43 for the Kewaunee Nuclear Power Plant (KNPP) to the extent held by Madison Gas & Electric Company (MG&E) to Wisconsin Public Service Corporation (WPSC), as approved by Order of the Commission dated September 20, 2001.

Date of issuance: September 27, 2001.

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

Amendment No.: 159.

Facility Operating License No. DPR–43: Amendment revised the Operating License and Technical Specifications.

Date of initial notice in **Federal Register:** July 27, 2001 (66 FR 39214). The August 3, 2001, supplement was within the scope of the initial application as originally noticed. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 20, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: August 8, 2000.

Brief description of amendments: The amendments revised the requirements for the containment isolation valves in the hydrogen/oxygen analyzer containment penetrations. The related safety evaluation also provided approval of an associated request to use closed system boundary valves that do not completely meet the guidance described in the Standard Review Plan, Section 6.2.4, ''Containment Isolation System.''

Date of issuance: September 28, 2001.

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

Amendment Nos.: 195 & 170.

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

Date of initial notice in **Federal Register:** June 12, 2001 (66 FR 31713). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: October 12, 2000, as supplemented April 9, 2001.

Brief description of amendment: This amendment changes the Technical Specifications (TSs) associated with the drywell vacuum breakers and the suppression pool vacuum breakers to provide consistency between the Hope Creek TSs and the improved standard TSs (NUREG–1433).

Date of issuance: October 3, 2001.

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

Amendment No.: 133.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in **Federal Register:** November 29, 2000 (65 FR 71137). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket No. 50–354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: May 17, 2001, as supplemented on August 6, August 17, and September 12, 2001

Brief description of amendment: The amendment revises the Technical Specifications to permit an increase in the allowable leak rate for the Main Steam Isolation Valves (MSIVs) and to delete the MSIV Sealing System. These changes are based on the use of an alternate source term and the guidance provided in Regulatory Guide 1.183, ''Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.''

Date of issuance: October 3, 2001.

Effective date: As of the date of issuance, and shall be implemented during Refueling Outage 10, currently scheduled to commence in October 2001.

Amendment No.: 134.

Facility Operating License No. NPF–57: This amendment revised the TSs.

Date of initial notice in **Federal Register:** June 27, 2001 (66 FR 34288). The letters dated August 6, August 17, and September 12, 2001, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 3, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: September 26, 2000, as supplemented on October 6, 2000, and May 21, 2001.

Brief description of amendments: The amendments modify the Salem Technical Specifications by increasing the as-found setpoint tolerance for the Pressurizer Safety Valves from ±1% to ±3%; increasing the as-found setpoint tolerance for the Main Steam Safety Valves (MSSV) from ±1% to ±3%; changing the required actions for inoperable MSSVs; and removing specifications and references related to plant operation with three Reactor Coolant System loops.

Date of issuance: September 19, 2001

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

Amendment Nos.: 244 and 225

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 15, 2000 (65 FR 69065), as superseded on August 8, 2001 (66 FR 41624).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 19, 2001.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: May 31, 2000, as supplemented on August 2, 2001.

Brief description of amendments: The amendments modify the Salem Technical Specifications (TSs) Surveillance Requirements for: (1) The Control Room Envelope Air Conditioning System (CREACS), (2) the Auxiliary Building Ventilation System (ABVS), and (3) the Fuel Handling Building Ventilation System (FHVS). Salem TSs will now require the use of American Society for Testing and Materials (ASTM) D3803–1989, ''Standard Test Method for Nuclear-Grade Activated Carbon,'' as the test protocol to evaluate charcoal samples from the ABVS, CREACS, and FHVS.

Date of issuance: September 19, 2001.

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

Amendment Nos.: 245 and 226.

Facility Operating License Nos. DPR–70 and DPR–75: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 26, 2000 (65 FR 46014). The August 2, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 19, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendments: June 29, 2001 as supplemented by letter dated August 20, 2001.

Brief description of amendments: 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 for a single inoperable train of both the ECW and CREACUS from 7 days to 14 days.

Date of Issuance: October 4, 2001.

Effective date: October 4, 2001, to be implemented within 30 days of issuance

Amendment Nos.: Unit 2–181; Unit 3–172

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

Date of initial notice in **Federal Register:** August 22, 2001 (66 FR 44175). The August 20, 2001 supplemental letter provided additional clarifying information, did not expand the scope of the proposed amendment as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 2001.

No significant hazards consideration comments received: No.

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

Date of amendments request: June 29, 2000, as supplemented August 31, 2001.

Brief description of amendments: The amendments revise design bases in the Final Safety Analysis Report. The change adds a description of the methodology Southern Nuclear Operating Company uses to determine what systems and components need to be protected from tornado missiles.

Date of issuance: September 26, 2001.

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

Amendment Nos.: 150 and 142.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Updated Final Safety Analysis Report.

Date of initial notice in **Federal Register**: August 9, 2000 (65 FR 48758).

The supplement dated August 31, 2001, provided clarifying information that did not change the scope of the June 29, 2000, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 26, 2001.

No significant hazards consideration comments received: No.

TXU Electric, Docket Nos. 50–445 and 50–446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: October 4, 2000, as supplemented by letters dated April 30, June 18, and July 18, 2001.

Brief description of amendments: The amendments revise the Technical Specifications to increase the spent fuel storage capacity from 2,026 to 3,373 fuel assemblies in the spent fuel pool.

Date of issuance: October 2, 2001.

Effective date: As of the date of issuance and shall be implemented no later than January 31, 2002.

Amendment Nos.: 87/87.

Facility Operating License Nos. NPF–87 and NPF–89: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register**: December 4, 2000 (65 FR 75737).

The April 30, June 18, and July 18, 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 October 2, 2001.

No significant hazards consideration comments received: No.

Vermont Yankee Nuclear Power Corporation, Docket No. 50–271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: April 17, 2001.

Brief description of amendment: The amendment removes unnecessary details for certain secondary post-accident monitoring instrumentation from Technical Specification Table 3.2.6.

Date of Issuance: October 2, 2001.

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

Amendment No.: 204.

Facility Operating License No. DPR–28: Amendment revised the Technical Specifications.

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

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 2, 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 23, 2001 (CO 01–0013).

Brief description of amendment: The amendment deletes (1) certain license conditions from Facility Operating License No. NPF–42, and (2) reporting requirements in Table 5.5.9–2, ''Steam Generator Tube Inspection,'' in Section 5.5.9, ''Steam Generator (SG) Tube Surveillance Program,'' of the technical specifications. License Conditions 2.C.(4), and 2.C.(6) through 2.C.(14), Section 2.F, and Attachments 2 and 3 to Facility Operating License No. NPF–42 are deleted, and the list of the attachments and appendices to Facility Operating License No. NPF–42 is revised to reflect the deletion of the attachments. The reporting requirements deleted in Table 5.5.9–2 duplicate requirements in 10 CFR 50.72.

Date of Issuance: September 24, 2001.

Effective date: September 24, 2001, and shall be implemented within 60 days from the date of issuance.

Amendment No.: 141.

Facility Operating License No. NPF–42: The amendment revised the operating license and the Technical Specifications.

Date of initial notice in **Federal Register***:* May 2, 2001 (66 FR 22035).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 24, 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: September 15, 2000, and supplements dated October 3, 2000, and September 13, 2001.

Brief description of amendment: The amendment revises footnotes (b) and (c) of Table 1.1–1, ''Modes,'' and adds a program plan to Section 5.5, ''Programs and Manuals,'' of the Wolf Creek Generating Station Technical Specifications. The amendment will allow the plant to operate at full power with one closure bolt less than fully tensioned for one operating cycle.

Date of issuance: September 27, 2001.

Effective date: September 27, 2001, to be implemented within 60 days from the date of issuance.

Amendment No.: 142.

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

Date of initial notice in **Federal Register***:* October 4, 2000 (65 FR 59227).

The supplements dated October 3, 2000, and September 13, 2001, provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 10th of October 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–25957 Filed 10–16–01; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Nonforeign Area Cost-of-Living Allowances Price and Background Surveys; Revised Collection; Comment Request

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995, the Office of Personnel Management (OPM) seeks comments on its intention to request reinstatement of two information collections whose approval period has expired. OPM has revised the two information collections to conform to the settlement agreement in the lawsuit *Caraballo, et al.* v. *United States*, No. 1997–0027 (D.V.I), August 17, 2000. OPM uses the two collections—a price survey and a background survey—to gather data to be used in determining cost-of-living allowances for certain Federal employees in Alaska, Hawaii, Guam and the Commonwealth of the Northern Mariana Islands, Puerto Rico, and the U.S. Virgin Islands. The price survey will be conducted in selected areas generally on an annual basis. The background survey will be conducted annually on a limited basis in preparation for each of the price surveys.

DATES: Submit comments on or before December 17, 2001.

ADDRESSES: *Comments:* Send or deliver comments to Donald J. Winstead, Assistant Director for Compensation Administration, Workforce Compensation and Performance Service, Office of Personnel Management, Room 7H31, 1900 E Street NW., Washington, DC 20415–8200; fax: (202) 606–4264, or email: cola@opm.gov. *Copies:* For copies of this proposal, contact Mary Beth Smith-Toomey at (202) 606–8358 or email: mbtoomey@opm.gov.

FOR FURTHER INFORMATION CONTACT: Kurt M. Springmann, (202) 606–2838.

SUPPLEMENTARY INFORMATION: Office of Management and Budget (OMB) approval of the Nonforeign Area Cost-of-Living Allowance Price Survey and Background Survey expired on August 31, 2001. OPM plans to request OMB approval for an additional 3 years and is seeking comments prior to submitting the collections to OMB for review. As set out in OMB regulations at 5 CFR 1320.8(d)(1), comments are requested to—

- Evaluate whether the surveys are necessary and have practical utility;