N–640) are based on the 1989 edition of the ASME Code.

Environmental Impacts of the Proposed Action

The NRC has completed its evaluation of the proposed action and concludes granting the exemption would provide an adequate margin of safety against brittle failure of the Byron and Braidwood reactor vessels. The proposed action (*i.e.*, granting the exemption) will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released off site, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential non-radiological impacts, the proposed action does not have a potential to affect any historic sites. It does not affect non-radiological plant effluents and has no other environmental impact. Therefore, there are no significant non-radiological environmental impacts associated with the proposed action.

Accordingly, the NRC concludes that there are no significant environmental impacts associated with the proposed action.

Environmental Impacts of the Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (*i.e.,* the ''no-action'' alternative). Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any different resource than those previously considered in the Final Environmental Statement for the Byron and Braidwood stations dated April 1982 and June 1984 respectively.

Agencies and Persons Consulted

On June 22, 2001, the staff consulted with the Illinois State official, Mr. Frank Niziolek of the Illinois Department of Nuclear Safety, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated July 5, 2000, as supplemented by letter dated December 8, 2000. 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 e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 19th day of July 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. 01–18520 Filed 7–24–01; 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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 2, 2001 through July 13, 2001. The last biweekly notice was published on July 11, 2001 (66 FR 36335).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By August 24, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, http://www.nrc.gov (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: June 7, 2001.

Description of amendment request: The proposed amendment request would revise the requirement for the Senior Manager-Operations to hold a Senior Reactor Operator license.

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

The proposed change to TS [Technical Specification] 6.2.2.2.j revises the requirement concerning the Operations management position that must hold an SRO [Senior Reactor Operator] license. At least one of the Operations Managers or the Senior Manager-Operations will continue to meet NRC requirements for maintaining an SRO license. The training, qualification, and experience requirements for Operations management personnel will continue to satisfy ANSI/ANS [American National Standards Institute/American Nuclear Society] 3.1–1978 as required by TS 6.3.1. This change does not involve any physical modifications to plant structures, systems, or components (SSC), or the manner in which SSCs are operated, maintained, modified, tested, or inspected. As the proposed change is administrative in nature, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change to TS 6.2.2.2.j revises the requirement concerning the Operations management position that must hold an SRO license. At least one of the Operations Managers or the Senior Manager-Operations will continue to meet NRC requirements for maintaining an SRO license. The training, qualification and experience requirements for Operations management personnel will continue to satisfy ANSI/ANS 3.1–1978 as required by TS 6.3.1. This change does not involve any physical modifications to SSCs, or the manner in which SSCs are operated, maintained, modified, tested, or inspected. As the proposed change is administrative in nature, operation of the facility in accordance with 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 amendment involve a significant reduction in a margin of safety?

The proposed change to TS 6.2.2.2.j revises the requirement concerning the Operations management position that must hold an SRO license. At least one of the Operations Managers or the Senior Manager-Operations will continue to meet NRC requirements for maintaining an SRO license. If the Senior Manager-Operations does not hold an SRO license, then an Operations Manager must hold an SRO license. This individual will be qualified to fill the Senior Manager-Operations position and have the same management authority over licensed operators as the Senior Manager-Operations. In addition, administrative procedures will ensure that there is always an individual holding a current SRO license within Operations management. The training, qualification and experience requirements for Operations management personnel will continue to satisfy ANSI/ANS 3.1–1978 as required by TS 6.3.1. This change does not involve any physical modifications to SSCs, or the manner in which SSCs are operated, maintained, modified, tested, or inspected. As the proposed change is administrative in nature, operation of the facility in accordance with 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: Kevin P. Gallen, Morgan, Lewis & Bockius, LLP, 1800 M Street, NW., Washington, DC 20036–5869.

NRC Section Chief: Richard P. Correia, Acting.

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

Date of amendment request: March 29, 2001, as supplemented by letter dated June 27, 2001.

Description of amendment request: The proposed amendment revises the pressure-temperature (P–T) limits of Technical Specification (TS) 3.1.2 for the Three Mile Island Nuclear Station, Unit 1 (TMI–1). The proposed amendment will revise the heatup, cooldown, and inservice hydrostatic test limitations, and the respective heatup and cooldown rates for the reactor coolant system (RCS). The service period for the new P–T limits will be for a maximum of 29 effective full power years. The related Bases are also revised. Sections 3.1.2.4 and 3.1.2.5 of the TSs are revised to remove reference to Sections V.B and V.C of Appendix G, of 10 CFR Part 50, as these sections no longer exist in the regulations. The proposed amendment also revises TS Figures 3.1–1 and 3.1–2 to permit TMI–1 to be operated during low temperature conditions with two reactor coolant pumps in operation in a single loop.

The proposed amendment also revises TS Section 3.1.12, ''Low Temperature Overpressure Protection (LTOP)'' setpoints. The RCS Power-Operated Relief Valve (PORV) low setpoint is being revised to 552 psig as a result of the P–T limit changes, which is the error-adjusted maximum setpoint. The enable temperature for the PORV setpoint (TS 3.1.12.2) and the LTOP setpoint (TS 3.1.12.1) is revised to 329 degrees Farenheit to be consistent with the new P–T bases. Section 3.1.12 is also revised to reorganize and clarify the LTOP system protection parameters and applicable conditions. Reference to nominal setpoint pressure values which do not affect the specified maximum and minimum setpoint values has been deleted. The related TS 3.1.12 Bases is revised to reflect the above changes to limits and setpoints. The Table of Contents page ii is revised to reflect the changes to Section 3.1.12 and to correct a previously issued typographical error in the listed titles of Sections 3.4.1 and 3.4.2.

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

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

These proposed Technical Specification changes were developed utilizing the procedures of ASME [American Society for Mechanical Engineers] [Boiler and Pressure Vessel Code (Code), Section] XI, Appendix G, in conjunction with Code Cases N–588 and N–640. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins of safety which ensure that failure of a reactor vessel will not occur.

The proposed changes do not impact the capability of the reactor coolant pressure boundary (i.e., no change in operating pressure, materials, seismic loading, etc.) and therefore, do not increase the potential for the occurrence of a loss of coolant accident (LOCA). The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, or construction standards.

The probability of any design basis accident (DBA) is not affected by this change, nor are the consequences of any DBA affected by this change. The proposed Pressure-Temperature (P–T) limits, Low Temperature Overpressure (LTOP) limits and setpoints, and allowable operating reactor coolant pump combinations are not considered to be an initiator or contributor to any accident analysis addressed in the TMI Unit 1 UFSAR [updated final safety analysis report].

The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Radiological off-site exposures from normal operation and operational transients, and faults of moderate frequency do not exceed the guidelines of 10 CFR [Part] 100. In addition, the proposed changes do not affect any fission product barrier. The revised PORV LTOP setpoint is established to protect [the] reactor coolant pressure boundary. The changes do not degrade or prevent the response of the PORV or safety-related systems to previously evaluated accidents. In addition, the changes do not alter any assumption previously made in the mitigation of the radiological consequences of an accident previously evaluated.

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

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

The proposed license amendment revises the TMI Unit 1 reactor vessel P–T limits, LTOP limits and setpoints, and allowable operating reactor coolant pump combinations. Compliance with 10 CFR 50 Appendix G, includes utilization of ASME XI, Appendix G, as modified by Code Cases N–588 and N–640 to meet the underlying intent of the regulations. The criteria of 10 CFR 50.61 remains satisfied, thus ensuring an adequate margin of safety for potential thermal shock events. The proposed limits are developed utilizing NRC-approved methodology and conservatively account for material property changes as required by regulation. The design basis event related to this change is nonductile failure of the reactor coolant pressure boundary. The proposed amendment provides assurance of protection against nonductile failure of the reactor coolant pressure boundary for operation of 29 Effective Full Power Years (EFPY) and is unrelated to the possibility of creating a new or different kind of accident. The proposed amendment does not introduce any new systems or components, or create any new component failure modes. Sufficient pressure margin is maintained to accommodate the proposed change to the allowable operating reactor coolant pump combinations.

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

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

The proposed Technical Specification (TS) changes were developed utilizing the procedures of ASME XI, Appendix G, in conjunction with Code Cases N–588 and N–640. Usage of these procedures provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides safety limits and margins of safety which ensure that failure of a reactor vessel will not occur.

No plant safety limits, set points, or design parameters are adversely affected. The fuel, fuel cladding, and Reactor Coolant System are not impacted.

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

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

Attorney for licensee: Edward J. Cullen, Jr., Esq., PECO Energy Company, 2301 Market Street, S23–1, Philadelphia, PA 19103.

NRC Section Chief: Richard P. Correia (Acting).

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

Date of amendments request: June 26, 2001.

Description of amendments request: The proposed amendments would revise the Technical Specifications to support a modification that would install a digital Power Range Neutron Monitoring (PRNM) system. The modification would supersede plant modifications previously installed in support of Carolina Power & Light Company's implementation of Enhanced Option I–A, and will allow full implementation of the Boiling Water Reactor Owners Group (BWROG) Option III Reactor Stability Long-Term Solution.

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

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

The proposed change will replace the currently installed and NRC approved Enhanced Option I–A long-term stability solution, which prohibits operation in areas with the potential for instability, with an NRC approved Option III long-term stability solution. The PRNM hardware meets the General Design Criteria (GDC) 10 and 12 requirements by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel Minimum Critical Power Ratio (MCPR) Safety Limit. The accident probability will not change since the instability is suppressed prior to exceeding the MCPR Safety Limit, the solution has defense-in-depth features, and is of robust design. In addition, the PRNM system does not interact with equipment whose failure could cause an accident, and compliance is retained for regulatory criteria established for PRNM system and associated plant equipment. Scram setpoints in the PRNM system will be established so that analytical limits are met. The reliability of the new system will meet or exceed that of the existing system and, as a result, the scram reliability will be equal to or better than the existing system. No new challenges to safety-related equipment will result from the PRNM system.

Proper operation of the PRNM system does not affect any fission product barrier or Engineered Safety Feature. Thus, the proposed change cannot change the consequences of any accident previously evaluated. As stated above, the PRNM system meets the requirements of GDC 10 and 12 by automatically detecting and suppressing design basis thermal-hydraulic oscillations prior to exceeding the fuel MCPR Safety Limit.

Based on the above, the operation of the new PRNM system and replacement of the currently installed Enhanced Option I–A stability solution with the Option III Oscillation Power Range Monitor (OPRM) function will not increase the probability or consequences of an accident previously evaluated.

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

The components of the PRNM system will be supplied to equivalent or better design and qualification criteria than is currently required for the plant. Equipment that could be affected by the PRNM system has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, system interaction, or equipment failure mode was identified. Therefore, the PRNM system will not adversely affect plant equipment.

The current plant design using the Enhanced Option I–A long-term stability solution depends on prohibited operating regions with an automatic scram if the exclusion region of the power/flow map is entered and an automatic rod block if the restricted region of the power/flow map is entered. The current design also relies on operator action to manually scram the plant if automatic monitoring of neutron flux through the period based detection system (PBDS) provides an instability alarm when in a region that has a potential for instability. The modification implementing PRNM replaces these automatic and manual requirements with a fully automatic detect and suppress capability to assure that instability events that occur will be terminated before the MCPR Safety Limit is exceeded. The ''scram and rod block enforced'' restrictions on the operating region are relaxed. Potential failures in the OPRM Upscale function could result in either failure to take the required mitigating action or an unintended reactor scram, which are the same potential effects of failure of the currently installed Enhanced Option I–A functions.

The PRNM modification and associated changes to the Technical Specifications involve equipment that is designed to detect the symptoms of certain events or accidents and initiate mitigating actions. The worst [case] failure of the equipment involved in the modification is a failure to initiate mitigating action (i.e., scram or rod block), but no failure can cause an accident of a new or different kind than any previously evaluated.

Based on the above, the proposed change 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 current safety analyses assume that the existing Enhanced Option I–A related Technical Specification requirements are adequate to prevent an instability event. PBDS is provided as part of the design to detect and suppress an instability event as a defense-in-depth feature. As a result, there is currently no impact on the MCPR Safety Limit identified for an instability event.

The Option III OPRM trip function is being implemented to fully automate the detection, via direct measurement of neutron flux, and subsequent suppression, via scram, of an instability event prior to exceeding the MCPR Safety Limit. Other OPRM trip features (i.e., Growth and Amplitude Algorithms) are provided as part of a robust design and defense-in-depth feature for unanticipated oscillations. Currently, the MCPR Safety Limit is not challenged by an instability event since the event is prevented by automatic means or mitigated by automatic and manual means via the Enhanced Option I–A functions. In both methods the margin of safety associated with the MCPR Safety Limit is maintained.

Other changes such as setpoint revisions, removing the Average Power Range Monitor Downscale function from the Reactor Protection System trip logic, removing the number of operable Local Power Range Monitors from the automatic trip logic, and lengthening the Surveillance Requirement frequencies are shown to be acceptable, as documented in licensing topical report (LTR) NEDC–32410P–A, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC–PRNM) Retrofit Plus Option III Stability Trip Function,'' October 1995, and LTR NEDC–32410P–A Supplement 1, ''Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC–PRNM) Retrofit Plus Option III Stability Trip Function,'' November 1997. Both of these LTRs have been reviewed and approved by the NRC.

Based on the above, the proposed change 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: William D. Johnson, Vice President and Corporate Secretary, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Section Chief: Patrick M. Madden, Acting.

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

Date of amendment request: May 7, 2001, as supplemented on June 29, 2001.

Description of amendment request: The proposed amendment revises Technical Specification (TS) 3.4.3 and the associated Surveillance Requirements (SR) to eliminate the pressurizer water volume value in the specification and change ''volume'' to ''level'' in TS 3.4.3 and SR 4.4.3.1.

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

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

The proposed changes do not affect operations of the Reactor Coolant System (RCS) components. The proposed change is administrative in nature in that it deletes a value from the TS that is not used as a control limit since it cannot be monitored directly. Instead, pressurizer level is used as the control parameter and level can be monitored. The volume specified in the current TS is redundant information to the level limit in the specification. The specification is made consistent with the Improved Technical Specifications [ITS] with this change. The ITS only identify a limit for percent pressurizer level. No change to the HNP TS for the pressurizer level value is being proposed.

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

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

The proposed changes do not involve new plant components or procedures, but only removes a value for volume in the pressurizer which is essentially redundant to the percent level indication and not a directly monitorable parameter for plant operation. These changes are administrative in nature and do not place SSCs [structures, systems, and components] in conditions outside of their design basis. There is no revision to operating setpoints or conditions.

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

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

The proposed changes to the pressurizer level TS and associated bases only remove unnecessary information from the specification. The information is not needed for plant operation and control. The deletion of this information represents an administrative change only since no change to the maximum level setpoint or operational limit is being made. The effect of this change is to make the plant TS consistent with the current ITS with no change to the margin of safety as described in the TS.

Therefore, the proposed change does not involve 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 amendment request involves no significant hazards consideration.

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

NRC Section Chief: Patrick M. Madden, Acting.

Dominion Nuclear Connecticut, Inc., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 11, 2001, as supplemented June 14, 2001.

Description of amendment request: The proposed amendment would update the list of documents describing the analytical methods used to determine the core operating limits specified in Technical Specification (TS) 6.9.1.8b. Specifically, these changes would update the documents describing the analytical methods used in the current Small Break Loss of Coolant Accident analysis (SBLOCA), setpoint methodology, and non-LOCA methodology. In addition, the revision number and the date of documents listed in TS 6.9.1.8b would be deleted, in a manner consistent with that approved by the NRC in Standard Technical Specification Change Traveler TSTF–363.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in their April 11, 2001, application. However, the NRC staff found that the licensee's no significant hazards consideration was not fully supported. In response to the staff's request, the licensee submitted a revised no significant hazards consideration on June 14, 2001, which is presented below:

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

The proposed change in document 6 and the deletion of document 7 of Technical Specification 6.9.1.8b are made to identify the most recent, Nuclear Regulatory Commission (NRC) approved, model used in Small Break Loss of Coolant Accident (SBLOCA) applications. This methodology meets the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K. This change has no impact on plant equipment operation. Since the change only affects the SBLOCA analysis, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

The proposed change in document 15 (renumbered 14) of Technical Specification 6.9.1.8b is made to identify the most recent, NRC approved, setpoint methodology for Combustion Engineering type reactors. This change has no impact on plant equipment operation. The proposed change does not revise any setpoints assumed in the accident analyses. Therefore, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

The proposed change to add a new document as 6.9.1.8b.15 is required to identify the most recent Non-LOCA methodology to be used in the Millstone Unit No. 2 Non-LOCA analysis. The use of this methodology will demonstrate that the acceptance criteria for Non-LOCA events are met. This change has no impact on plant equipment operation. The change does not affect the acceptance criteria for Non-LOCA accident[s]. Therefore, it cannot affect the likelihood or consequences of accidents. Therefore, this change will not increase the probability or consequences of an accident previously evaluated.

Deleting the revision number and the date from the documents contained in sections 6.9.1.8b.1 through 6.9.1.8b.15 has no impact on the actual analytical methods used to determine the core operating limits, nor does it have impact on the calculations performed for current or future reloads. This change is administrative in nature. This change has no impact on plant equipment operation nor does it 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 changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. These changes do not introduce any new failure modes. Therefore, the proposed changes 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.

The proposed changes have no impact on plant equipment operation. The proposed changes do not revise any setpoints assumed in the analyses and do not affect the acceptance criteria for Non-LOCA accidents. 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, Connecticut 06385.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 31, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification 5.5.6.b, ''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.

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 current Bases Control Program allows the licensee to make changes to the Technical Specification Bases that do not modify the Technical Specification requirements and which are allowed without prior NRC approval via 10 CFR 50.59. The proposed change does not modify these requirements and is administrative in nature. The revised change modifies the wording of the Bases Control Program to be consistent with the revised 10 CFR 50.59 program. The evaluation requirements of 10 CFR 50.59 will ensure that changes to the Technical Specification Bases will not result in more than a minimal increase in the probability or consequences of an accident without NRC prior review and approval. Therefore, this change does 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 analyzed?

The current Bases Control Program allows the licensee to make changes to the Technical Specification Bases that do not modify the Technical Specification requirements and which are allowed without prior NRC approval via 10 CFR 50.59. The proposed change does not modify these requirements and is administrative in nature. The revised change modifies the wording of the Bases Control Program to be consistent with the revised 10 CFR 50.59 program. The evaluation requirements of 10 CFR 50.59 will ensure that changes to the Technical Specification Bases will not result in a new or different kind of accident than any previously evaluated in the final safety analysis report without NRC prior review and approval. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The current Bases Control Program allows the licensee to make changes to the Technical Specification Bases that do not modify the Technical Specification requirements and which are allowed without prior NRC approval via 10 CFR 50.59. The proposed change does not modify these requirements and is administrative in nature. The revised change modifies the wording of the Bases Control Program to be consistent with the revised 10 CFR 50.59 program. The evaluation requirements of 10 CFR 50.59 will ensure that changes to the Technical Specification Bases will not result in significant reduction in the margin of safety without NRC prior review and approval. This change is administrative in nature based on the amending of 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: J. M. Fulton, Esquire, Assistant General Counsel, Pilgrim Nuclear Power Station, 600 Rocky Hill Road, Plymouth, Massachusetts, 02360–5599.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: May 30, 2001.

Description of amendment request: The proposed amendments would change the Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1.3 and add two new SRs, SR 3.6.1.1.4 and SR 3.6.1.1.5, covering the testing of Suppression Chamber-Drywell Vacuum Breakers. The proposed changes will decrease the frequency of the Drywell-to-Suppression Chamber bypass leakage test while maintaining the current leakage test frequency for the Suppression Chamber-Drywell Vacuum Breakers, and establish new leakage acceptance criteria for the Suppression Chamber-Drywell Vacuum Breakers when the valves are tested individually. The proposed TS changes are similar to TS changes approved for Susquehanna Steam Electric Station on September 6, 1996.

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

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

The proposed changes modify Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.1.3 and add two new SRs, SR 3.6.1.1.4 and SR 3.6.1.1.5. The proposed changes will decrease the frequency for the Drywell-to-Suppression Chamber bypass leakage test while maintaining the current leakage testing frequency for the Suppression Chamber-Drywell Vacuum Breakers, and establish new leakage acceptance criteria for the Suppression Chamber-Drywell Vacuum Breakers when the valves are tested individually.

The performance of a Drywell-to-Suppression Chamber bypass leakage test or Suppression Chamber-Drywell Vacuum Breaker leakage test is not a precursor to any accident previously evaluated. Thus, the proposed changes to the performance of the leakage tests do not have any effect on the probability of an accident previously evaluated.

The performance of a Drywell-to-Suppression Chamber bypass leakage test or a Suppression Chamber-Drywell Vacuum Breaker test does provide assurance that the containment will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the proposed changes do 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 an accident from any accident previously evaluated?

The proposed changes to SR 3.6.1.1.3, SR 3.6.1.1.4, and SR 3.6.1.1.5 do not affect the assumed accident performance of any LaSalle County Station structure, system or component previously evaluated. The proposed changes do not introduce any new modes of system operation or failure mechanisms.

Therefore, the proposed changes do 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 current frequency associated with a Drywell-to-Suppression Chamber bypass leakage test in SR 3.6.1.1.3 is 24 months or 12 months if two consecutive tests fail and continues at this frequency until two consecutive tests pass. The proposed SR change will modify the leakage test frequency to be consistent with the Primary Containment Leakage Rate Testing Program for Type A Tests, or 48 months following one test failure or 24 months if two consecutive tests fail and continues at this frequency until two consecutive tests pass. The proposed change in SR 3.6.1.1.3 frequency is acceptable as the results from previous tests show that the measured Drywell-to-Suppression Chamber bypass leakage at the current TS frequency has been a small percentage of the allowable leakage. Acceptability is further demonstrated by the design requirements applied to the primary containment components and other periodically performed primary containment inspections.

The proposed SR 3.6.1.1.4 will establish a leakage test frequency of 24 months for each Suppression Chamber-Drywell Vacuum Breaker except when the leakage test of SR 3.6.1.1.3 has been performed within 24 months. SR 3.6.1.1.4 specifies a leakage limit for each Suppression Chamber-Drywell Vacuum Breaker of less than or equal to 12% of the bypass leakage limit of TS 3.6.1.1.3. The proposed SR 3.6.1.1.5 will establish a total leakage limit of less than or equal to 30% of the bypass leakage limit of SR 3.6.1.1.3 when the Suppression Chamber-Drywell Vacuum Breakers are tested in accordance with SR 3.6.1.1.4. The proposed changes to establish leakage limits for the Suppression Chamber-Drywell Vacuum Breakers are acceptable as demonstrated by the results from previous Suppression Chamber-Drywell Vacuum Breaker leakage tests that show the measured leakage has been a small percentage of the allowable leakage.

Thus, 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 requested amendments involve no significant hazards consideration.

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

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 23, 2001.

Description of amendment request: The proposed change would delete the loose parts monitoring system (LPMS) and the associated Technical Specifications (TSs) and Bases currently contained in the Limerick Generating Station, Units 1 and 2 Technical Specifications. The licensee bases its proposal to delete the LPMS on the conclusions of the Boiling Water Reactor Owners' Group Topical Report NEDC–32975P, ''Regulatory Relaxation for BWR Loose Parts Monitoring Systems''.

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

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

This Technical Specification (TS) Change Request will delete the Loose Parts Monitoring System and the associated Technical Specifications and Bases currently contained in the Limerick Generating Station (LGS), Units 1 and 2, Technical Specifications. The Loose Parts Monitoring System (LPMS) is not an accident initiating system. The LPMS was designed in conformance with Regulatory Guide 1.133 (''Loose-Parts Detection Program for the Primary System of Light-Water-Cooled Reactors,'' Revision 1, May 1981), to detect and alarm for loose parts in the reactor coolant system. A secondary function of the system is to assist the operators in locating the detected loose parts. The LPMS is used for information purposes only and is not a safety-related system. The operators do not rely solely on this system or information provided by this system for the performance of any safety-related action. Review of the Updated Final Safety Analysis (UFSAR) indicates that this system is not relied upon by other systems for input or data. This is a monitoring system that does not perform any automatic or control functions, and is not relied upon for any accident or transient evaluation. The removal of the LPMS from operation will not increase the need for operator intervention or increase operator burden to support any system used to mitigate an accident under normal or off normal conditions. Therefore, the proposed changes will not significantly increase the probability of an accident previously evaluated.

The removal of the LPMS will not change or degrade the physical barriers or systems designed to contain radiation, and will have no affect on the on-site or off-site radiological conditions. Therefore, the proposed TS changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

This TS Change Request will delete the Loose Parts Monitoring System and the associated Technical Specifications and

Bases currently contained in the LGS, Units 1 and 2, Technical Specifications. Removal of this system will not create a new mode of operation of the plant. The LPMS is a nonsafety-related monitoring system. The proposed changes do not create a system-level failure mode different than those that already exist. In addition, there are no operation or failure modes of the LPMS that are accident initiators. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This TS Change Request does not affect any safety limits or analytical limits. Also there are no changes to accident or transient core thermal hydraulic conditions, or fuel or reactor coolant boundary design limits, as a result of these proposed changes. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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: March 28, 2001.

Description of amendment request: The proposed amendment request for BVPS–1 would increase the limits for boron concentration in the Refueling Water Storage Tank (RWST) and in the Reactor Coolant System (RCS) accumulators. This proposed license amendment would also revise the limits and associated surveillance requirements on boron concentration in the Boron Injection Tank (BIT) to be consistent with the limits specified for the RWST. This proposed amendment would revise the RCS minimum boron concentration limit for Mode 6 to make it consistent with the RWST boron concentration limit.

The increase in the boron concentration limits in the RWST and Accumulators is needed to address higher reactor core reactivity levels associated with core operation with higher plant capacity factors. The RCS boron concentration limit in Mode 6 during refueling needs to be revised whenever the RWST/Accumulator minimum boron concentration limit is adjusted, for consistency. Boron concentration above the upper limit of the RWST are not needed in the BIT in order to satisfy applicable safety analyses. Therefore, revising the boron concentration limits removes the need to maintain associated temperature controls and their associated surveillance requirements on the BIT. The Note at the bottom of Technical Specification (TS) 3/4.5.4.1.1 is being deleted since N–1 loop operation during Modes 1, 2 and 3 (when this Specification is applicable) is not permitted by TS 3/4.4.1.4.1.

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 change to the BVPS Unit 1 RWST, Accumulators, BIT and in the RCS during Mode 6 will maintain the safety analyses results in Chapter 14 of the BVPS Unit 1 UFSAR [Updated Final Safety Analysis Report] as bounding values for all Loss of Coolant Accident (LOCA) and non-LOCA design basis accidents. The proposed changes do not invalidate the RWST, accumulators or BIT's ability to meet its design bases.

Increased boron concentration limits for the RWST, Accumulators, BIT and RCS in Mode 6 will not increase the consequences of an accident previously analyzed. The increased boron concentration limits reduce the time to switchover from cold leg to hot leg recirculation, which will prevent boron precipitation in the reactor vessel following a LOCA. The post-LOCA long term core cooling minimum boron requirements have been determined to continue to be adequate to ensure adequate post-LOCA shutdown margin. The post-LOCA containment sump and containment spray pH remain within the limits specified in the UFSAR. All other transients either were not impacted or were made less severe as a result of the increased boron concentrations.

The deletion of the Note in Technical Specification 3/4.5.4.1.1 does not alter the safety analyses as evaluated in the UFSAR since N–1 operation is currently prohibited by Technical Specification 3/4.4.1.4.1. With the reduced upper limit on boron concentration in the BIT, the controls on temperature for the BIT are eliminated since boron precipitation is precluded above freezing.

Therefore, this change will not increase the probability of occurrence of a postulated accident or the consequences of an accident previously evaluated since the change would continue to comply with the current BVPS Unit 1 licensing basis as it relates to the peak cladding temperature criteria of 10 CFR Part 50, Appendix K and the dose limits of GDC 19 and 10 CFR Part 100.

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

The proposed increase in boron concentration does not add new or different equipment to the facility. The proposed Technical Specification changes also do not alter the manner in which plant equipment is being operated. Although the increased boron concentration requires procedure changes to ensure that cold leg to hot leg recirculation after a LOCA occurs quicker, there are no changes to the methods utilized to respond to plant events. The proposed Technical Specification changes do not alter instrument or control setpoints that initiate protective or mitigative actions. These increased boron concentration limits are conservative and do not alter the RCS or Emergency Core Cooling System's ability to perform their design bases.

The deletion of the Note in Technical Specification 3/4.5.4.1.1 does not alter the safety analyses as evaluated in the UFSAR since N–1 operation is currently prohibited by Technical Specification 3/4.4.1.4.1. With the reduced upper limit on boron concentration in the BIT, the controls on temperature for the BIT are eliminated since boron precipitation is precluded above freezing.

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

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

The LOCA considerations, including Peak Cladding Temperature calculations, containment sump and spray pH requirements, boron solubility requirements, cold shutdown boration requirements, post-LOCA long term core cooling minimum boron requirements, hot leg recirculation switchover requirements, post-LOCA hydrogen generation requirements, and radiological requirements have been evaluated and determined to be acceptable. The acceptance criteria of all non-LOCA design basis accidents continue to be met.

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not adversely affect the ability of systems, structures or components important to the mitigation and control of design bases accident conditions within the facility. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be maintained in a shutdown or refueling conditions for extended periods of time.

Based upon the above evaluations, the [change does not involve a significant reduction in the margin of safety [* * *]

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

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

NRC Section Chief: Richard Correia, Acting.

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

Date of amendment request: June 12, 2001.

Description of amendment request: The requested amendment would revise the Technical Specifications (TSs) by changing Surveillance 4.4.10 to incorporate alternative reactor coolant pump (RCP) flywheel inspection requirements and would make various administrative wording changes to TSs 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3.

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

In accordance with 10 CFR 50.92, North Atlantic has concluded that the proposed changes do not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed changes do not involve a SHC is as follows:

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

This proposed revision to TS Surveillance 4.4.10, incorporates alternative reactor coolant pump flywheel inspection requirements into TS Surveillance 4.4.10 based on Topical Report WCAP–14535A. WCAP–14535A provided a technical basis for the elimination of inspection requirements for reactor coolant pump flywheels based on industry data. The industry data indicated that no indications that would affect the integrity of flywheels were revealed during 729 examinations of 217 flywheels at 57 plants (including Seabrook Station). The NRC, during their review and approval of the WCAP required continued inspections on a ten-year interval to protect against events and degradation that were not anticipated and had not been considered in the WCAP analysis. The proposed alternate inspection requirements are consistent with the conclusions of an NRC review and generic approval of Topical Report WCAP–14535A. Thus, it is concluded that the proposed revision to TS Surveillance 4.4.10 does not significantly increase the probability of an accident. Additionally, the performance of reactor coolant pump flywheel surveillances does not increase the consequence of an accident previously evaluated.

The proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. In addition, these proposed changes do not affect the manner in which the plant responds in normal operation, transient or accident conditions, nor do they change procedures related to operation of the plant. The proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not alter or prevent the ability of structures, systems and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). These proposed changes are administrative in nature and only update the Operation [sic] License.

The proposed changes to TS 4.4.10, 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 are administrative in nature and only update the Seabrook Station Operating License. These proposed changes do not affect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

Therefore, it is concluded that these proposed revisions to TS 4.4.10, 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

This proposed revision to TS Surveillance 4.4.10 does not change the operation or the design basis of any plant system or component during normal or accident conditions. The proposed change incorporates alternate inspection requirements for the reactor coolant pump flywheels, which were generically approved by the NRC for use by licensees. This change does not include any physical changes to the plant. The proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes.

The proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 are administrative in nature and only update the Seabrook Station Operating License. These proposed changes do not modify the facility, nor do they modify the manner in which the plant will be operated, nor do they affect the plant's response to normal, transient or accident conditions. The proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not introduce a new mode of plant operation. The plant's design and design basis are not revised and the current safety analyses will remain in effect and the plant will continue to be operated in accordance with the existing Technical Specifications.

Thus, these proposed revisions to TS 4.4.10, 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This proposed revision to TS Surveillance 4.4.10 incorporates alternative reactor coolant pump flywheel inspection requirements into TS Surveillance 4.4.10 that are consistent with the conclusions of an NRC review and generic approval of Topical Report WCAP–14535A. The current inspection requirements of TS Surveillance 4.4.10 and the NRC review of WCAP–14535A were both based on the recommendations of Regulatory Guide 1.14. The proposed changes do not change the function or operation of plant equipment or affect the response of that equipment if it is called upon to operate. The performance capability of the reactor coolant pumps will not be affected. Reactor coolant pump reliability and availability will be unaffected by implementation of the proposed changes.

The proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 are administrative in nature and only update the Seabrook Station Operating License. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the TSs are not revised. Neither the plant design, nor its method of operation, are revised by these proposed changes. Finally, the proposed changes to TS 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not change the physical design or the operation of the plant.

Thus, it is concluded that these proposed revisions to TS 4.4.10, 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not involve a significant reduction in a margin of safety.

Based on the above evaluation, North Atlantic concludes that the proposed changes to TS 4.4.10, 6.4.1.7.b, 6.4.2.2.d, and 6.4.2.3 do not constitute a significant hazard.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: June 18, 2001.

Description of amendment request: The proposed amendment would revise the reference point for reactor vessel level instrumentation specifications to use instrument ''zero'' instead of ''top of active fuel'; simplify the Safety Limits and Limiting Safety System Settings to eliminate specifications that are unnecessary, outdated, or redundant to other Technical Specifications (TSs); change the reactor coolant system pressure Safety Limit from 1335 psig to 1332 psig to correct a minor calculation error; and make corresponding TS Bases changes.

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

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

The requested changes are administrative in nature in that they change instrumentation reference points, reformat sections to conform to current NRC guidance, or correct minor errors.

One change involves a small conservative reduction in the reactor coolant system pressure limit. This change corrects a long standing minor discrepancy in this numerical limit.

Another change eliminates the extra 12 inches above the top of active fuel currently specified in the reactor water level Safety Limit. It is sufficient to require that all active fuel is covered by water to satisfy the objective of the Safety Limit and assure the integrity of the fuel cladding.

None of these changes affect the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their radiological consequences are not adversely affected.

Therefore, the probability or consequences of an accident previously evaluated are not affected.

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 involve a change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Accordingly, no new failure modes have been created for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes.

Therefore the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

One change involves a small conservative reduction in the reactor coolant system pressure limit. This change corrects a long standing minor discrepancy in the derivation of the numerical value of this limit of less than 0.3%. The correction is conservative.

Another change eliminates the extra 12 inches above the top of active fuel currently specified in the reactor water level Safety Limit. The additional 12 inches of water does not significantly contribute to fuel cooling under plant conditions for which the Safety Limit would be applicable. While the change in reactor water level represents a less restrictive limit, the proposed numerical value still ensures an adequate margin for core cooling and provides an adequate margin for effective action. The benefits gained from achievement of uniformity with the reactor water level Safety Limit established by the NRC for plants similar to Monticello outweigh any negative aspects of this change.

The remainder of the requested changes are administrative in nature or correct minor errors.

Therefore, a significant reduction in the margin of safety is not involved in the proposed changes.

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: Claudia M. Craig.

Omaha Public Power District, Docket No. 50–285, Fort Calhoun Station, Unit No. 1, Washington County, Nebraska

Date of amendment request: June 18, 2001.

Description of amendment request: The proposed amendment will delete Technical Specification (TS) Sections 5.14.3 and 5.14.4, ''Post-Accident Radiological Sampling and Monitoring,'' requirements to maintain a Post Accident Sampling System (PASS). Licensees were generally required to implement PASS upgrades as a result of NUREG–0737, ''Clarification of TMI [Three Mile Island] Action Plan Requirements,'' and Regulatory Guide 1.97, Revision 3, ''Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident.'' Implementation of these upgrades was an outcome of the NRC's lessons learned from the accident that occurred at TMI Unit 2. Requirements related to PASS were imposed by Order for many facilities and were added to or included in the TS for nuclear power reactors currently licensed to operate. Lessons learned and improvements implemented over the last 20 years have shown that the information obtained from PASS can be readily obtained through other means or is of little use in the assessment and mitigation of accident conditions.

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in a license amendment application in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated June 18, 2001.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Attorney for licensee: James R. Curtiss, Esq., Winston & Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: May 24, 2001.

Description of amendment request: The Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications (TS) Sections 4.2.2.2.e and g and 4.2.2.4.e and g would be changed to adopt a revised methodology that relocates the Heat Flux Hot Channel Factor $F_Q(z)$ penalty for increasing $F_Q(z)$ versus burnup to a table in the Core Operating Limits Report (COLR). Also proposed is an increase in the $F_Q(z)$ surveillance region to be consistent with the current core design and to provide assurance that the peak $F_Q(z)$ is monitored and evaluated near end of core life.

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 to the measurement and evaluation of the maximum $F_Q(z)$ will provide conservative limits for assuring the plant is operated in a safe and consistent manner. No changes are being made that could initiate an accident. The consequences of accidents previously evaluated are unaffected by these proposed changes as no change to equipment response or accident mitigation capabilities (including assessment capabilities) has occurred. The proposed changes have no impact on the principal safety barriers of the plant.

Therefore, the change does 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 decrease the size of the core region that is excluded from the evaluation of peak $F_Q(z)$ and relocate penalties from the TS to the COLR per an approved methodology. No new accident scenarios, failure mechanisms or limiting single failures are introduced as the result of this proposed change. This change does not challenge the integrity or performance of any safety-related system.

Therefore, the possibility of a new or different kind of accident is not created.

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

No.

The proposed change relocates the penalties associated with measuring $F_Q(z)$ and decreases the size of the core regions excluded from the TS required surveillance for peak $F_Q(z)$. There is no effect on the availability, operability, or performance of the safety-related systems, structures, or components. The margin of safety associated with the acceptance criteria for any accident is unchanged. All surveillances will be performed at their required frequencies and with the same acceptance criteria, which assures the plant conditions prior to transients, events, and accidents [remain] within the conditions assumed in the safety analyses.

The Bases of the TS are founded in part on the ability of the regulatory criteria being satisfied assuming limiting conditions for operation for various systems. Conformance to the regulatory criteria for operation with $F^M_Q(z)$ penalty factor relocation and the $F^M_Q(z)$ exclusion region changes is demonstrated, and the regulatory limits are not exceeded. Therefore, there is no significant reduction in the margin of safety resulting from the proposed changes.

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: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: June 19, 2001.

Description of amendment request: This proposed change supports the inclusion of the newer versions of the process rack circuit boards into the response time testing elimination population. These versions of the cards were not included in the original Failure Modes and Effect Analysis performed for WCAP–14036–P–A, Revision 1.

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

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

This change to the Technical Specifications (TS) does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same [Reactor Trip System] RTS and [Engineered Safety Features Actuation System] ESFAS instrumentation is being used; the time response allocations/modeling assumptions in the Final Safety Analysis Report (FSAR) Chapter 15 analyses are still the same; only the method of verifying the time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed change will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR.

Therefore, the proposed change does 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?

This change does not alter the performance of process protection racks, Nuclear Instrumentation, and/or logic systems used in the plant protection systems. These systems will still have response time verified by test before being placed in operational service. Changing the method of periodically verifying instrument[s] for these systems (assuring equipment operability) from response time testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these systems will continue and may be used to detect degradation that could cause the response time to exceed the total allowance. The total time response allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Implementation of the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for the process protection racks, Nuclear Instrumentation, and logic systems is modified to allow the use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will continue to be performed and may be used to detect any degradation which might cause the system response time to exceed the total allowance. The total response time allowance for each function bounds all degradation that cannot be detected by periodic surveillance. Based on the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety.

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

Attorney for licensee: Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: June 27, 2001.

Description of amendment request: The proposed amendment would revise the Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications Surveillance Requirement (SR) 3.8.1.13 frequency from once every 18 months (with a maximum of 22.5 months including the 25% grace period of SR 3.0.2) to once every 24 months (for a maximum of 30 months including the 25% grace period of SR 3.0.2). The proposed change would allow SR 3.8.1.13 to be performed following the Diesel Generator inspection/maintenance, which is performed at 24-month intervals in accordance with manufacturer recommendations. Similarly, the frequency of SR 3.8.1.14 would be revised from once every 18 months to once every 24 months. The proposed change would allow SR 3.8.1.14 to be performed following SR 3.8.1.13.

Bases for proposed no significant hazards consideration determination: As required by 10 CFR Part 50 the licensee has provided its analysis of the requested revision of the surveillance as an issue of no significant hazards consideration and is presented below:

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

No. The surveillance intervals associated with SRs 3.8.1.13 and 3.8.1.14 have no bearing on the likelihood of any of the initiating events assumed for any of the accidents previously evaluated. Therefore, increasing the intervals for SRs 3.8.1.13 and 3.8.1.14 do not involve a significant increase in the probability of any accident previously evaluated. The operability of the emergency diesel generators (DGs) will continue to be demonstrated by all of the other surveillance requirements associated with TS Limiting Condition for Operation (LCO) 3.8.1 which are not affected by the proposed change. Endurance and margin will continue to be demonstrated by SR 3.8.1.13, and hot restart functional capability will continue to be demonstrated by SR 3.8.1.14. The only difference will be the increased surveillance intervals, which have been shown to have a minimal impact on safety in accordance with Generic Letter 91–04. Therefore, the DGs will remain capable of performing their safety function as assumed in the accident analyses, and the proposed changes do not involve a significant increase in the consequences of any accident previously evaluated.

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

No. The proposed changes do not introduce any new equipment or create new failure modes for existing equipment. No new limiting single failure is created, and plant operation will not be altered. The DGs will remain capable of performing their safety function as assumed in the safety analyses. No other safety-related or important-to-safety equipment is affected by the proposed changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any previously evaluated.

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

No. The operability of the emergency diesel generators (DGs) will continue to be demonstrated by all of the other surveillance requirements associated with TS Limiting Condition to Operation (LCO) 3.8.1 which are not affected by the proposed changes. Endurance and margin and hot restart functional capability will continue to be demonstrated by SRs 3.8.1.13 and 3.8.1.14, respectively. The only difference will be the increased intervals, which have been shown to have a minimal impact on safety in accordance with Generic Letter 91–04. The proposed changes are consistent with current regulatory guidance and licensing actions for increasing TS surveillance intervals to accommodate operating cycles that have been extended to 24 months. 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: Mr. Arthur H. Domby, Troutman Sanders, NationsBank Plaza, Suite 5200, 600 Peachtree Street, NE., Atlanta, Georgia 30308–2216.

NRC Section Chief: Richard L. Emch, Jr.

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

Date of amendment request: May 14, 2001.

Description of amendment request: The proposed amendment would revise Technical Specification Section 3.3.5, ''Loss of Power (LOP) Diesel Generator Start Instrumentation,'' to increase the time delay setting of the 6.9 kV Shutdown Board degraded voltage relays from a nominal 6 seconds to 10 seconds. This change will provide the plant with operating margin by allowing additional time for the Class 1E Auxiliary Power System to react to projected voltage transients on the offsite grid. This will aid in preventing unnecessary challenges to the WBN Class 1E power supply due to spurious relay actuations which result in automatic transfer from the WBN preferred offsite power supply to the emergency standby diesel generators.

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 degraded voltage protection relays and associated time delay relays provided for each of the four 6.9 kV Shutdown Boards act to mitigate the consequences of previously analyzed accidents by detecting a sustained undervoltage condition, isolating the safety buses from offsite power, and starting the associated diesel generators. This safety function and logic of the degraded voltage relay circuits remains unchanged. The revised time delay setpoint will allow automatic load tap changers on CSSTs [Common Station Service Transformers] C and D additional time to react to voltage transients on the offsite grid. This will aid in preventing unnecessary relay actuation and isolation from offsite power sources, which in turn will reduce the probability of a loss of offsite power to the unit due to voltages transients on the offsite grid. The additional four second time delay does not introduce any new constraints that would prevent safety equipment from performing its designed function. The only impact to equipment previously evaluated is an increase in the exposure to a degraded voltage condition (for the loads fed from the 6.9 kV Shutdown Boards) for a duration of an additional four seconds. However, the required safety-related equipment would continue to operate throughout the 10 second delay. The proposed change will not contribute to any radiological dose during an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The 6.9 kV Shutdown Power System will continue to function as specified in the design basis. The Class 1E loads supplied by the 6.9 kV Shutdown Boards will continue to be available to perform their intended safety function during the degraded voltage condition. The affected 6.9 kV Shutdown Boards will satisfactorily recover the voltage either by: (1) Stabilization of the offsite power grid if the degraded voltage condition is resolved within 10 seconds, or (2) transfer to emergency power if condition is present at the end of 10 seconds. There are no changes in the credible failure modes of the 6.9 kV Shutdown Boards (including the degraded voltage relays and timers) from those identified and evaluated previously in the FSAR [Final Safety Analysis Report]. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The ability of Class 1E loads fed from the 6.9 kV Shutdown Boards to perform their safety function is not compromised by this change. The lower boundary dropout and the upper reset setpoint of the degraded voltage relays remains unchanged. Increasing the delay time from 6 to 10 seconds will not change the voltage recovery profile. Analyses has shown that all motors will have adequate voltage to accelerate to their rated speed within their required times and therefore, there is no impact on operating equipment. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Patrick M. Madden, Acting.

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

Date of amendment request: May 14, 2001.

Description of amendment request: The proposed amendment would revise the Watts Bar Nuclear Plant Unit 1 Technical Specifications (TS) and TS Bases to eliminate the requirements associated with core alterations from those limiting condition of operations (LCOs) that provide safety functions to mitigate the consequences of a fuel handling accident. The affected specifications are LCOs 3.3.6, 3.3.7, 3.7.10, 3.7.11, 3.9.4, 3.9.7, and 3.9.8.

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

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

The proposed revision eliminates requirements associated with core alterations for specifications that are intended to mitigate the consequences of a fuel handling accident (FHA). These functions will not impact accident generation because their function is to support mitigation of accidents and they are not considered to be the source of a postulated accident. The removal of these actions affects functions that are not necessary during core alterations because postulated events during these activities do not have the potential to result in major fuel cladding damage like that assumed for an FHA. Therefore, there is no adverse impact to nuclear safety by eliminating core alteration requirements for specifications that provide for the mitigation of an FHA.

The proposed revision does not adversely alter any plant equipment or operating practices; therefore, the probability of an accident is not significantly increased. In addition, the consequences of an accident are not significantly increased by eliminating core alteration requirements for specifications that only support the mitigation of FHAs. This is based on sufficient safety function capabilities being available for the mitigation of an FHA or other potential events that could occur during core alteration activities.

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

The proposed allowance to eliminate core alteration requirements for FHA related specifications will not adversely alter plant functions or equipment operating practices. The proposed elimination of core alteration requirements will not impact accident generation because these functions provide for FHA mitigation and are not postulated to be an initiator of postulated accidents. Therefore, since plant functions and equipment are not adversely affected and the availability of FHA mitigation functions do not contribute to the initiation of postulated accidents, the proposed revision will not create a new or different kind of accident.

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

The elimination of core alteration requirements for specifications that provide mitigation functions for FHAs will not affect the ability of these functions to perform as necessary. This is based on postulated events during core alteration not having the potential to result in fuel cladding damage that is assumed for the FHA and therefore, not requiring functions necessary to mitigate the FHA event. The proposed revision will continue to provide acceptable provisions for activities that could result in an FHA or events postulated during core alterations to maintain the necessary margin of safety.

Therefore, the margin of safety provided by specifications required for the mitigation of

FHAs is not significantly reduced by the proposed allowance to eliminate core alterations requirements.

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

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

NRC Section Chief: Patrick M. Madden, Acting.

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

Date of amendment request: June 21, 2001.

Description of amendment request: This amendment request proposes to revise the control rod block instrumentation requirements contained in Technical Specification (TS) 2.1.B, Figure 2.1.1, and Tables 3.2.5 and 4.2.5. Some of the control rod block trip functions are being relocated to the Vermont Yankee Technical Requirements Manual and some of the requirements for the retained trip functions are clarified. Two trip functions are added to the TSs.

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

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The relocated trip functions are not assumed as initial conditions for, nor are they credited in the mitigation of, any design basis accident or transient previously evaluated. Since reactor operation with these revised and relocated Specifications is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events, nor the analyzed mitigation of design basis accident or transient events.

More stringent requirements that ensure operability of equipment and purely administrative changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

None of the proposed changes affects any parameters or conditions that could contribute to the initiation of any accident. No new accident modes are created since plant operation is unchanged. No safety-related equipment or safety functions are altered as a result of these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

This change does not impact plant equipment design or operation, and there are no changes being made to safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. Since the changes have no effect on any safety analysis assumptions or initial conditions, the margins of safety in the safety analyses are maintained. In addition, administrative changes that do not change technical requirements or meaning, and the imposition of more stringent requirements to ensure operability, have no negative impact on margins of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. David R. Lewis, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037–1128.

NRC Section Chief: James W. Clifford.

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.

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

Date of application for amendment: February 14, 2000, as supplemented on May 3, 2001.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to correct various editorial errors and make other administrative changes. Specifically, the amendment makes administrative changes that revise: (a) Tables 3.6–1 and 4.4–1 to correct listing and editorial errors, (b) TS 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv), (c) Figures 3.10–2 through 3.10–6 to remove these figures, (d) Table 4.1–1 to reflect change in level indication components, (e) TS 4.19.B and 6.14.1.1 to correct editorial errors, (f) TS 6.12.1 to reflect an organizational title change, and (g) TS 6.13.2 to correct a typographical error. In the May 3 letter, the licensee requested that the proposed changes to TS 6.12.1 regarding references to the current sections of 10 CFR Part 20 be withdrawn.

Date of issuance: July 5, 2001.

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

Amendment No.: 216.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications. *Date of initial notice in* **Federal Register:** February 21, 2001 (66 FR 11055).

The May 3, 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 July 5, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 21, 2000.

Brief description of amendment: The amendment approves a change to the licensing basis to allow a 121-second delay in the timing of the release of fission products following design-basis accidents.

Date of issuance: July 12, 2001.

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

Amendment No.: 143.

Facility Operating License No. NPF–43: Amendment revised the Updated Safety Analysis Report.

Date of initial notice in **Federal Register:** December 27, 2000 (65 FR 81914) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 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: December 20, 1999, as supplemented by letters dated November 29, 2000, and April 6, May 7, and June 7, 2001.

Brief description of amendment: The amendment changes River Bend Station (RBS) Technical Specification (TS) 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs),'' to allow the Inclined Fuel Transfer System (IFTS) primary containment isolation blind flange to be removed during MODES 1, 2, or 3. In its application, the RBS licensee stated that, with the blind flange removed and certain restrictions and administrative controls in place, the IFTS penetration would continue to be provided through implementation of these additional controls.

Date of issuance: July 3, 2001.

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

Amendment No.: 116.

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

Date of initial notice in **Federal Register:** January 26, 2000 (65 FR 4273).

The November 29, 2000, and April 6, May 7, and June 7, 2001, supplemental letters provided information that was within the scope of the original **Federal Register** notice and did not change the staff's initial no significant hazards consideration determination.

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

Brief description of amendment: The amendment deletes Technical Specification (TS) 6.8.4.d, ''Post-accident Sampling, for Waterford Steam Electric Station, Unit 3, and thereby eliminates the requirement to have and maintain the post-accident sampling system.

Date of issuance: July 3, 2001.

Effective date: As of the date of issuance and shall be implemented by February 28, 2002.

Amendment No.: 172.

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

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29353).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 26, 2000, as supplemented January 17, 2001, and April 17, 2001.

Brief description of amendments: Revised the Technical Specifications (TSs) Index to delete reference to the BASES since, in accordance with 10 CFR 50.36(a), the BASES are not a part of the TSs required by 10 CFR 50.36, and to include a ''Technical Specification (TS) Bases Control Program'' in the Administrative Controls Section of the TS.

Date of Issuance: July 12, 2001.

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

Amendment Nos.: 176 and 117.

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

Date of initial notice in **Federal Register:** October 4, 2000 (65 FR 59222). The letters dated January 17, 2001, and April 17, 2001, contained clarifying information that did not affect the original proposed no significant hazards determination, or expand the scope of the request as noticed.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 12, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 5, 2001, as revised by letter dated March 30, 2001

Brief description of amendment: The amendment changes Technical Specification (TS) Section 5.5.12, ''Programs and Manuals—Technical Specifications (TS) Bases Control Program,'' in accordance with Nuclear Energy Institute TS Task Force (TSTF) Standard TS Change Traveler, TSTF–364, ''Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59,'' Revision 0.

Date of issuance: July 9, 2001.

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

Amendment No.: 204.

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

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

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

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 14, 2001.

Brief description of amendment: The amendment makes minor revisions in the Ginna Station Improved Technical Specifications (ITS) format to allow for maintaining, viewing, and publishing them with a different software package. The amendment also includes a revision to ITS Section 5.5.13, ''Technical Specifications (TS) Bases Control Program,'' to provide consistency with the changes to 10 CFR 50.59 as published in (64 FR 53852 dated October 4, 1999).

Date of issuance: June 26, 2001.

Effective date: June 26, 2001.

Amendment No.: 80.

Facility Operating License No. DPR–18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 21, 2001 (66 FR 15929).

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 12, 2001.

Brief description of amendments: Revised the Technical Specifications (TS) and associated Bases to change the methodology and frequency for sampling the ice condenser ice bed (stored ice) and adds a new TS and associated bases to address sampling requirements for all ice additions to the ice bed.

Date of issuance: July 12, 2001.

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

Amendment Nos.: 269 and 259.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22033).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 12, 2001.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland this 17th day of July 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–18324 Filed 7–24–01; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Proposed Collection; Comment Request

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

Extension:

Regulation A and Forms 1–A and 2–A, OMB Control No. 3235–0286, SEC File No. 270–110.

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

Regulation A provides an exemption from registration under the Securities Act for certain limited securities offerings by issuers who do not otherwise file reports with the Commission. Form 1–A is an offering statement filed under Regulation A. Form 2–A. Form 2–A is used to report sales and use of proceeds in Regulation A offerings. Approximately 186 issuers file Forms 1–A and 2–A. It is estimated that Form 1–A takes 608 hours to prepare, Form 2–A takes 12 hours to prepare and Regulation A takes one administrative hour to review for a total of 621 hours per response. The total annual burden hours are 115,506. It is estimated that 75% of the 115,506 total burden hours (86,630 burden hours) would be prepared by the company.

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

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

Dated: July 18, 2001.

Margaret H. McFarland,

Deputy Secretary.

[FR Doc. 01–18517 Filed 7–24–01; 8:45 am]

BILLING CODE 8010–01–M

SECURITIES AND EXCHANGE COMMISSION

[Investment Company Act Release No. 25068; 812–12422]

Nationwide Mutual Funds and Villanova Mutual Fund Capital Trust

July 19, 2001.

AGENCY: Securities and Exchange Commission (''Commission'').

ACTION: Notice of an application under section 17(b) of the Investment Company Act of 1940 (the ''Act'') for an exemption from section 17(a) of the Act.

SUMMARY OF APPLICATION: Applicants request an order to permit a series of Nationwide Mutual Funds (''Nationwide'') to acquire substantially all of the assets, net of liabilities, of another series of Nationwide (the ''Reorganization''). Because of certain affiliations, applicants may not rely on rule 17a–8 under the Act.

FILING DATE: The application was filed on January 30, 2001. Applicants have agreed to file an amendment to the application during the notice period, the substance of which is reflected in this notice.

HEARING OR NOTIFICATION OF HEARING: An order granting the application will be issued unless the Commission orders a hearing. Interested persons may request a hearing by writing to the Commission's Secretary and serving applicants with copies of the request, personally or by mail. Hearing requests should be received by the Commission by 5:30 p.m. on August 13, 2001, and should be accompanied by proof of service on applicants, in the form of an affidavit or, for lawyers, a certificate of service. Hearing requests should state the nature of the writer's interest, the reason for the request, and the issues contested. Persons may request notification of a hearing by writing to the Commission's Secretary.

ADDRESSES: Secretary, Commission 450 5th Street, NW., Washington, DC 20549–0609. Applicants, c/o Elizabeth A. Davin, Esq., Nationwide Mutual Funds, One Nationwide Plaza, 1–35–16, Columbus, Ohio 43215.

FOR FURTHER INFORMATION CONTACT: Bruce R. MacNeil, Senior Counsel, at