Morrissey by November 6 at the address indicated.

Organizations or individuals may also submit statements for the record without testifying. Twenty (20) copies of such statements should be sent to the Executive Secretary of the Advisory Council at the above address. Papers will be accepted and included in the record of the meeting if received on or before November 6, 2001.

Signed at Washington, DC this 25th day of October, 2001.

**Ann L. Combs,**

*Assistant Secretary, Pension and Welfare Benefits Administration.*

[FR Doc. 01–27359 Filed 10–30–01; 8:45 am]

**BILLING CODE 4510–29–M**

## NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

**[NOTICE (01–136)]**

### NASA Advisory Council (NAC), Space Science Advisory Committee Meeting

**AGENCY:** National Aeronautics and Space Administration.

**ACTION:** Notice of meeting.

**SUMMARY:** In accordance with the Federal Advisory Committee Act, Pub. L. 92–463, as amended, the National Aeronautics and Space Administration announces a forthcoming meeting of the NASA Advisory Council, Space Science Advisory Committee.

**DATES:** Wednesday, December 5, 2001, 8:30 a.m. to 5:30 p.m., and Thursday, December 6, 2001, 8:30 a.m. to 5:30 p.m.

**ADDRESSES:** Hilton Cocoa Beach Oceanfront, 1550 North Atlantic Avenue, Cocoa Beach, Florida 32931–3268.

**FOR FURTHER INFORMATION CONTACT:** Ms. Marian Norris, Code SB, National Aeronautics and Space Administration, Washington, DC 20546, 202/358–4452.

**SUPPLEMENTARY INFORMATION:** The meeting will be open to the public up to the capacity of the room. The agenda for the meeting includes the following:

—Associate Administrator's Program Status Report
—Division Managers' Reports
—Subcommittee Reports
—In-Space Propulsion
—Mars Exploration
—Strategic Planning Status
—Technology Programs Update
—GPRA Science Objectives Assessment

It is imperative that the meeting be held on these dates to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitor's register.

**Beth M. McCormick,**

*Advisory Committee Management Officer, National Aeronautics and Space Administration.*

[FR Doc. 01–27392 Filed 10–30–01; 8:45 am]

**BILLING CODE 7510–01–P**

## NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

**[Notice 01–137]**

### Aerospace Safety Advisory Panel (ASAP); Meeting

**AGENCY:** National Aeronautics and Space Administration.

**ACTION:** Notice of meeting.

**SUMMARY:** In accordance with the Federal Advisory Committee Act, Pub. L. 92–463, as amended, the National Aeronautics and Space Administration announces a forthcoming meeting of the Aerospace Safety Advisory Panel.

**DATES:** Friday, November 9, 2001, 8 a.m. to 11:30 p.m. Eastern Standard Time.

**ADDRESSES:** Radisson Resort At the Port, Cape Canaveral, 8701 Astronaut Blvd., Cape Canaveral, FL 32920. Martinique Room. Hotel phone number is (321) 784–0000.

**FOR FURTHER INFORMATION CONTACT:** Mr. David M. Lengyel, Aerospace Safety Advisory Panel Executive Director, Code Q–1, National Aeronautics and Space Administration, Washington, DC 20546, 202/358–0391, if you plan to attend.

**SUPPLEMENTARY INFORMATION:** This meeting will be open to the public up to the seating capacity of the room (40). The agenda for the meeting is to conduct deliberations on Calendar Year 2001 fact-finding activities and trip reports in preparation for the drafting of the Panel's Annual Report. It is imperative that the meeting be held on this date to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitors register.

**Beth M. McCormick,**

*Advisory Committee Management Officer, National Aeronautics and Space Administration.*

[FR Doc. 01–27393 Filed 10–30–01; 8:45 am]

**BILLING CODE 7510–01–P**

## NUCLEAR REGULATORY COMMISSION

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

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

### I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from October 9, 2001 through October 19, 2001. The last biweekly notice was published on October 17, 2001 (66 FR 52794).

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

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

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

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

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

By November 30, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC web site, *http://www.nrc.gov/NRC/ADAMS/index.html.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

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

*Docket Nos. 50–289, Three Mile Island Nuclear Station, Unit 1, Dauphin County, Pennsylvania*

*Date of amendment request:* August 1, 2001.

*Description of amendment request:* The requested changes to the technical specifications (TSs) propose to revise requirements that have been superceded based on licensed operator training programs being accredited by the Institute for Nuclear Power Operations (INPO), promulgation of the revised 10 CFR part 55, Operators' Licenses, and adoption of a systems approach to training as required by 10 CFR 50.120, Training and qualification of nuclear power plant personnel. The same changes were requested by Exelon Generation Company, LLC (Exelon) for Braidwood Station, Units 1 and 2; Byron Station, Units 1 and 2; Dresden Nuclear Power Station, Units 2 and 3; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed no significant hazards consideration for those plants is published elsewhere in the **Federal Register** under Exelon Generation Company, LLC.

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

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

There will be no changes to the procedures by which the operators operate the plants. There will be no changes to the systems, structures, or components in the plants.

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

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

There will be no changes to the procedures by which the operators operate the plants. There will be no changes to the systems, structures, or components in the plants.

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

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

There will be no change in the plants' systems, structures, or components, nor in the way in which they will be operated as a result of the proposed changes. Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

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

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

*NRC Section Chiefs:* Anthony J. Mendiola, Lakshminaras Raghavan.

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

*Date of amendment request:* July 31, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated July 31, 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:* Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* August 27, 2001.

*Description of amendment request:* The proposed amendment would revise the Technical Specification action and surveillance requirements associated with the containment air lock and expand the current guidance provided to address inoperable air lock components.

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

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

The proposed Technical Specification changes to revise the action and surveillance requirements associated with the containment air lock will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The containment air lock is not an accident initiator. The proposed changes will not revise the operability requirements (e.g., leakage limits) for the containment air lock. Proper operation of the containment air lock will still be verified. As a result, the design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report, and the consequences of the design basis accidents will remain the same. Therefore, the proposed changes 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 to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed 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 analyzed.

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

The proposed Technical Specification changes to revise the action and surveillance requirements associated with the containment air lock will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The operability requirements for the containment air lock have not been changed. The containment air lock will continue to function as assumed in the safety analysis. In addition, the proposed changes will not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. 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, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* August 28, 2001.

*Description of amendment request:* The proposed amendment would revise the Technical Specifications by removing the surveillance requirement that verifies the automatic opening features of the safety injection tank outlet isolation valves.

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

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

The proposed Technical Specification change to remove the surveillance requirement that verifies the automatic opening features of the safety injection tank outlet isolation valves will not cause an accident to occur since the safety injection tanks and associated isolation valves are not accident initiators. In addition, the proposed change will not alter the operation of the associated accident mitigation equipment. The operability requirement for the safety injection tank outlet isolation valves to be deenergized open when the safety injection tanks are required to be operable will not be affected, and outlet isolation valve position will still be verified periodically. As a result, the design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report, and the consequences of the design basis accidents will remain the same. Therefore, the proposed change will not increase the probability or consequences of an accident previously evaluated.

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

The proposed Technical Specification change does not impact any system or component that could cause an accident. The proposed change will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed change will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed change does not introduce any new failure modes. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously analyzed.

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

The proposed Technical Specification change to remove the surveillance requirement that verifies the automatic opening features of the safety injection tank outlet isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The proposed change will not revise the operability requirement for the safety injection tank outlet isolation valves to be deenergized open when the safety injection tanks are required to be operable. The safety injection tanks will continue to be able to mitigate the design basis accidents as assumed in the safety analysis. In addition, the proposed change will not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. Therefore, the proposed change will not result in a reduction in a margin of safety.

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

*Attorney for licensee:* Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* June 4, 2001.

*Description of amendment request:* The proposed amendment modifies the Millstone Nuclear Power Station, Unit No. 2 (MP2) and Unit No. 3 (MP3) Technical Specifications (TSs) to relocate selected MP2 and MP3 technical specifications related to the reactor coolant system to the respective Technical Requirements Manual (TRM), with the exception of MP3 Technical Specification section 4.4.10, which will be relocated to section 6 of MP3's TS. The Bases of the affected TSs will be modified to address the proposed 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. The NRC staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's analysis, which is based on the representation made by the licensee in the June 4, 2001, application, is presented below:

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

The proposed requirements remain the same except that the requirements will be relocated to the TRM. Since the proposed requirements are the same, this proposed change will not increase the probability or consequences of an accident previously evaluated.

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

Since the requirements remain the same, these proposed changes do not alter the way any system, structure, or component functions and do not alter the manner in which the plant is operated. The proposed 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.

Since the proposed changes are solely to relocate the existing requirements, it does not affect plant operation in any way. Therefore, the proposed change will not result in a reduction in a margin of safety.

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

*Attorney for licensee:* Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Hartford, CT 06141–5127.

*NRC Section Chief:* James W. Clifford.

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

*Date of amendment request:* July 31, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated July 31, 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:* Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

*NRC Section Chief:* James W. Clifford.

*Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York*

*Date of amendment request:* September 20, 2001.

*Description of amendment request:* The proposed amendment would revise Technical Specification (TS) 3.8, ''Refueling, Fuel Storage and Operations with the Reactor Vessel Head Bolts Less Than Fully Tensioned,'' TS Table 4.1–2, ''Frequencies for Sampling Tests,'' and TS Section 5.4, ''Fuel Storage,'' to allow the credit for soluble boron in the criticality analysis of the spent fuel pit (SFP). The revisions also incorporate changes to the SFP rack layout by dividing it into sub-regions and specifying requirements for fuel assembly burnup and soluble boron concentration for various loading configurations in these sub-regions.

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

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

Current TS contain minimum requirements for the SFP boron concentration. The actual boron concentration in the SFP has been maintained at a higher value. The proposed changes to the TS establish new boron concentration requirements for the SFP water that are consistent with the new criticality analysis. Since soluble boron has already been maintained in the SFP water and is currently required by the TS, the implementation of this new requirement will have no effect on the normal SFP operations and maintenance.

The presence of an increased requirement for soluble boron in the SFP water does not increase the probability of a fuel assembly drop accident in the SFP. The handling of the fuel assemblies in the SFP has always been performed in borated water. The criticality analysis shows the consequences of a fuel assembly drop accident in the SFP are not affected when considering the presence of soluble boron since the rack $k_{eff}$ remains $\leq$ 0.95.

Fuel assembly placement will continue to be controlled in accordance with approved fuel handling procedures and will be in accordance with TS spent fuel rack storage configuration limitations. The proposed SFP storage configuration limitations will be more complex but will be similar to those previously approved. Therefore, the new limitations will not significantly increase the probability of accident occurrence. There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel racks since the criticality analysis demonstrates that the SFP $k_{eff}$ will remain $\leq$ 0.95 following an accidental misloading.

There is no increase in the probability of the loss of normal cooling to the spent fuel pit water when considering the presence of soluble boron in the pit water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

Soluble boron requirements for mitigating reactivity effects due to increased pool temperatures are adequately met by the proposed increase in minimum TS soluble boron concentration. A negligible increase in the probability of a criticality accident due to increased pool temperature exists with the proposed TS changes, as the minimum soluble boron concentration will not change. The positive reactivity introduced as a result of the higher TS boron concentration effect on moderator reactivity coefficient will be sufficiently mitigated by the substantial margin to the amount actually required to maintain $k_{eff} \leq 0.95$.

Decreased fuel temperatures will increase the water density in the SFP, therefore increasing the thermal neutron flux, possibly causing an increase in reactivity. This density increase will increase the differential worth of the soluble boron but the excess soluble boron in the SFP is more than sufficient to offset any reactivity increase introduced by a temperature decrease.

Therefore, the proposed changes do 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.

Spent fuel handling accidents are not new or different types of accidents, they have been analyzed for the UFSAR [Updated Final Safety Analysis Report] and in Criticality Analysis reports associated with License Amendment 150 up to the nominal 5.0 $^{w}/_{o}$ $^{235}U$ [weight percent uranium-235] that is assumed for the proposed change.

A dilution of the SFP soluble boron has always been a possibility. However the boron dilution event previously had no consequences since boron was not previously credited. With the proposed TS, credit is taken for soluble boron. So a boron dilution has been evaluated as a possible new accident. The evaluation concluded a boron dilution accident was not credible, that processes were in place to detect and mitigate the possible events, and that, even if the SFP boron concentration was diluted to zero, criticality would not occur. Therefore, there would be no additional hazards if this request were approved.

There is no other change in the plant configuration or equipment design.

Therefore, the proposed change does not create a new accident initiator or precursor, or 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 TS changes proposed by this LAR [license amendment request] and the resulting spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. These limits are based on the plant specific criticality analysis and boron dilution analysis. The proposed TS changes rely upon known and predictable reactivity effects to ensure required criticality margins in the SFP.

While the criticality analysis utilizes credit for soluble boron, storage configurations have been defined using 95/95 $k_{eff}$ calculations to ensure the spent fuel rack $k_{eff}$ will be <1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions and to provide subcritical margin such that the SFP $k_{eff}$ is maintained ≤0.95.

The loss of substantial amounts of soluble boron from the SFP, which could lead to $k_{eff}$ exceeding 0.95, has been evaluated and shown to be not credible. An evaluation has been performed that shows that the dilution of the SFP boron concentration from 2000 ppm [parts per million] to 786 ppm is not credible. Also the spent fuel rack $k_{eff}$ will remain <1.0 with the SFP flooded with unborated water. These safety analyses demonstrate a level of safety comparable to the conservative criticality analysis approved for License Amendment 150 and show that the requirements of 10CFR50.68 are met.

The reactivity credit for additional poisons in the spent and fresh fuel assemblies increases the margin of safety in the SFP. No credit is taken for Boraflex in certain regions, when in reality some residual Boraflex does remain in these regions. In regions that do take credit for Boraflex, the amount of credit is conservative. These conservatisms add an increased safety margin. Predictions of the effective neutron multiplication factors have shown that, under the worst of scenarios, the SFP remains subcritical when conservative credit for future expected loss of Boraflex poison plates is considered.

The analysis show that the level of safety required by 10CFR50.68 is achieved for the IP2 [Indian Point 2] SFP with the proposed TS.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in [a] margin of safety.

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

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

*NRC Section Chief:* L. Raghavan, Acting.

*Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York*

*Date of amendment request:* September 20, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated September 20, 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 [Three Mile Island 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 [a] 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 [a] margin of safety.

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

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

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

*NRC Section Chief:* L. Raghavan, Acting.

*Entergy Nuclear Operations, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York*

*Date of amendment request:* September 20, 2001.

*Description of amendment request:* The proposed amendment would allow the one-time extension of the intervals for selected Technical Specification (TS) surveillance requirements (SRs) to enable the tests to be performed during the next refueling outage starting no later than November 19, 2002. Specifically, the surveillance interval would be extended for certain SRs associated with the volume control tank (VCT), residual heat removal system (RHR) , emergency diesel generators (EDGs), and shock suppressors (snubbers). In addition, the proposed amendment would: (1) Correct the channel functional test interval in Items 3 and 4 of TS Table 4.10–4 and Items 4 and 5 of Table 4–10–4, (2) delete alternate inspection requirements for the steam generator snubbers, and (3) remove the reference to a prior one-time extension of checks, calibrations and tests for certain instrument channels in TS Table 4.1–1 that is no longer applicable.

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

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

There is no change to the design, function, or capability of any plant structure, system, or component as a result of the proposed surveillance interval extensions. Hence there is no change in the probability of occurrence of an accident previously evaluated.

The proposed surveillance interval extensions do not affect the ability of any plant structure, system, or component to mitigate the consequences of any accident previously evaluated. The surveillance interval extensions do not alter or prevent the ability of the affected structures, systems, and components to perform their intended functions.

[VCT]

A statistical analysis of channel uncertainty for a proposed 31 month operating cycle has been performed. It confirms that the channel drift for the proposed 31 month interval is bounded by the existing drift allowance used in the current uncertainty calculations. Therefore, there is no expected decrease in reliability for the VCT level channel for the proposed 31 month operating cycle. Since there is no expected decrease in the reliability of the VCT level channels, the design safety functions of the VCT are not affected.

[RHR]

Since the past test data supports the integrity of the system and an extended standby period is not expected to affect any potential leak path, there is a reasonable expectation that the RHR and Safety Injection systems will continue to perform their intended safety functions without excessive leakage. It is concluded that a one-time extension of less than one month for the leakage test surveillance intervals will have minimal impact on the system reliability.

[EDG]

The identified anomalies with valve and filter operation for EDG 23 were evaluated and corrected and are not indicative of any inability of the machine to meet performance requirements. The anomalous adjustment affecting movement of the fuel control lever arm for EDG 22 was properly evaluated and eliminated as evidenced by subsequent successful testing. Therefore, the historical data together with the positive verification of the adequacy of corrective actions for previous test failures demonstrate that the EDGs have met the required performance criteria. Therefore the ability of the EDGs to mitigate accidents is not affected by this proposed change.

Failure of an EDG cannot, of itself, initiate an accident.

[Snubbers]

The TS functional testing program requires a sampling program that provides a 95% confidence level that 90–100% of the snubbers operate within acceptance limits. For each snubber failing the functional test an additional sample lot must be selected and tested to assure that the required confidence level is maintained. The past functional test history with very few functional test failures provides assurance that an extension in the surveillance will not result in increased snubber failures. In all cases, the functional test failures were thoroughly analyzed and appropriate action was taken to prevent recurrence. Subsequent testing resulted in all snubbers meeting their design requirements.

The operability of snubbers is not affected by the deletion of the allowance to separately group steam generator snubbers for the purposes of determining inspection intervals.

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

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

The proposed changes do not involve any physical design change or operational change to any plant system, structure or component. Thus a new failure mode is not introduced. Therefore, the proposed changes do not create a new accident initiator or precursor, or create the possibility of a new or different kind of accident from any accident previously evaluated.

[VCT]

The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the Updated Final Safety Analysis Report (UFSAR). The proposed change in the surveillance interval has been evaluated to have a negligible effect on the reliability of the existing instruments.

[RHR]

The proposed change does not involve the addition of any new or different type of equipment. Nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the UFSAR.

[EDG]

The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the UFSAR. Also, the increased surveillance interval (one-time only) will not adversely affect the reliability of the EDGs.

[Snubbers]

The proposed license amendment does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed change does not involve the addition of any new or different type of equipment, nor does it involve operating equipment required for safe operation of the facility in a manner that is different from that addressed in the UFSAR. Also, the increased surveillance interval (one-time only) will not adversely affect the snubbers.

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

As a result of these proposed surveillance interval extensions, there are no changes to IP2's design or to the IP2 TS safety limits, limiting safety system settings, or limiting conditions for operation. The only change is a change to the surveillance testing frequency for affected structures, systems, and components.

The proposed surveillance interval extensions have been evaluated to not significantly degrade the reliability of any existing system, structure, or component. Therefore, testing in accordance with the proposed test intervals continues to ensure that the necessary quality of affected structures, systems, and components is maintained, that IP2 operation will be within safety limits, and that the IP2 limiting conditions for operation will be met.

The proposed surveillance interval extensions do not adversely affect the ability of any IP2 structures, systems, or components to function when required to mitigate any accident or licensing basis event.

[VCT]

The proposed change in surveillance interval resulting from an increased operating cycle will not result in a channel statistical allowance that impacts any TS limit or any UFSAR requirement. Protective functions will continue to occur so that safety analysis limits are not exceeded.

Based on past rest results, the one-time extension of nine days does not involve a significant reduction in a margin of safety.

[RHR]

There is minimal risk that a surveillance interval extension of less than one month will increase leakage in the piping systems under review beyond the TS limits or that the system performance will be influenced. Past test data indicate that there was no impact on the margin imposed by the TS.

[EDG]

The functional test history indicates the functional test failures were the result of actions independent of actual EDG load performance. Apart from these anomalous actions, the record does not indicate a potential for failure to meet performance criteria. In all cases, the functional test failures were thoroughly analyzed and appropriate actions were taken to prevent recurrence.

Subsequent testing resulted in the EDG meeting its design requirements.

There is no reduction of margin indicated by the surveillance testing. The proposed change for a one-time extension of the test interval does not adversely affect the performance of any safety related system, component or structure and does not result in increased severity of any of the accidents considered in the UFSAR. Surveillance test results indicate no trend toward margin reduction.

[Snubbers]

The objective of the functional test is to provide a 95% confidence level that 90–100% of the snubbers operate within the specified acceptance limits. The review of past test history indicates that this objective was met at the time of the testing. There are no identified trends that would suggest that the same success rate would not be maintained over the requested extension period. The proposed license amendment does not involve a significant reduction in a margin of safety. The proposed change for a one-time extension of the test interval does not adversely affect the performance of any safety related system, component or structure and does not result in increased severity of any of the accidents considered in the UFSAR.

Therefore, the one-time extension of less than one month for the functional tests does not involve a significant reduction in a margin of safety.

The proposed deletion of the allowance to separately group steam generator snubbers for the purpose of determining inspection intervals does not affect the effectiveness of the surveillance requirements. The steam generator snubbers will still be inspected at the interval required by the TS.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in the margin of safety.

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

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

*NRC Section Chief:* L. Raghavan, Acting.

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

*Date of amendment request:* October 2, 2001.

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

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

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

The two degraded voltage protection relays that are provided on each of the 480 V safety buses act to mitigate the consequences of an accident by detecting a sustained undervoltage condition, isolating the safety buses from offsite power, and starting the associated emergency diesel generator (EDG). This safety function is unchanged by the proposed allowable voltage setting revisions. The revised settings for the degraded voltage protection relays will continue to provide the safety function of protecting the associated Class 1E equipment from the effects of a low voltage condition. The time delays remain within those assumed in the ANO–2 [Arkansas Nuclear One, Unit 2] safety analyses. Additionally, the revised allowable voltage settings will not result in any unnecessary isolation from the off-site power sources. The relocation of trip setpoint values to station surveillance procedures allows operational flexibility to account for additional margins, drifts, or uncertainties while ensuring that the relays are set to actuate within the acceptable range of allowable values denoted in the TSs. Since the proposed change does not adversely impact the mitigating function of the relays, the consequences of an accident previously evaluated remains unchanged.

The ANO–2 technical specifications will continue to require the 480 V bus degraded voltage functions to be surveillance tested at their present frequency without changing the modes in which the surveillance is required or the modes of applicability for these components. The technical specifications will continue to require the same actions as currently exist for the inoperability of one or more of the 480 V bus degraded voltage relays.

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

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

The proposed change introduces no new modes of plant operation or new plant configuration that could lead to a new or different kind of accident from any previously evaluated being introduced. The 480 V bus degraded voltage relays are required to operate upon detection of a sustained undervoltage condition to protect the Class 1E components from damage from low voltage by initiating transfer of the 4160 V safety bus power source to the EDG. This safety function remains unchanged by the proposed allowable voltage setting revisions, and the proposed values continue to provide the required actions consistent with the ANO–2 safety analysis.

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

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

The two degraded voltage relays located on each 480 V safety bus are provided to detect sustained undervoltage, isolate the safety buses, and start the EDGs. This safety function remains unchanged by the proposed revisions to the allowable values. The proposed changes to the allowable values for the degraded voltage relays incorporate channel uncertainties and calibration tolerances, while fully meeting their required safety functions of degraded voltage protection without resulting in undesired tripping of the offsite power source.

The slightly higher range of allowable values for the degraded voltage settings allows enhanced protection of the Class 1E components, but does not result in undesired tripping of the offsite power source for the analyzed grid minimum normal condition. In addition, the slight increase in the range of allowable values for the degraded voltage time delay remains well within the assumption of the accident analysis.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* October 2, 2001.

*Description of amendment request:* The proposed amendment would relocate the technical specification (TS) requirement that the reactor core be subcritical for a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool (SFP), to the technical requirements manual (TRM).

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

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

The accident of concern related to the proposed change is the fuel handling accident. This accident assumes a dropped fuel assembly. One of the assumptions made in the analysis is that fuel movement is delayed at least 100 hours after shutdown to allow for radioactive decay of the fission product inventory. TS 3.9.3.a provides this restriction. The analysis does not assume any further delay in fuel movement following the initial 100-hour decay period. The relocation of TS 3.9.3.b will not impact this assumption.

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

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

The proposed change relocates TS 3.9.3.b to the TRM. There are no changes to the design or operation of the facility proposed. Thus, there are no new or different kinds of accidents created. SFP cooling capability and the heat load generated by the movement of fuel into the SFP will continue to be evaluated under 10 CFR 50.59. The SFP cooling system includes two cooling pumps and one heat exchanger. In addition, several systems are available for makeup when needed. Under postulated accident conditions, when no pool cooling systems are operational, the maximum temperature at the inlet to the cells is assumed to be equal to the saturation temperature at atmospheric pressure or 212F [Fahrenheit] (allowed to boil). The proposed change does not increase the possibility of a complete loss of pool cooling.

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

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

The proposed change relocates TS 3.9.3.b to the TRM. Following relocation, any future changes to TRM 3.9.3.b will be assessed under the guidance of 10 CFR 50.59. The ANO [Arkansas Nuclear One] 50.59 process will provide an evaluation to ensure heat loads transferred will be within the cooling capacity of the service water system. Analyses will continue to demonstrate that even in the event of a loss of SFP cooling, the maximum temperature in the pool is such that design limits associated with assuring the integrity of the fuel cladding are satisfied.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* October 2, 2001.

*Description of amendment request:* The proposed amendment would change the technical specification definitions of response time for the reactor trip system (RTS) and for engineered safety features (ESFs) to allow use of either an allocated or a measured response time for select sensors in these two 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. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response time testing is not an initiator of any previously evaluated accident. The proposed change to the definition of RTS and ESF response time allows substitution of an allocated response time for selected sensors in lieu of measuring the sensor response time. The allocated response times adequately represent the response time of the components such that the safety systems utilizing these components will continue to perform their accident mitigation function as assumed in the safety analysis. Response time testing for the non-sensor portions of the channels will continue to use a series of sequential or overlapping test measurements.

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

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

The proposed change does not involve a physical change to the plant. Modifications will not be made to existing components nor will any new or different types of equipment be installed. The proposed change modifies the definitions for RTS and ESF response time and allows the substitution of an allocated response time in lieu of measured sensor response time for selected sensors. The response time assumed in the accident analysis for the non-sensor portions of the channels will continue to be verified using a series of sequential or overlapping test measurements. Appropriate actions will be taken to ensure overall channel response time remains within the times specified in the accident analysis.

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

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

The proposed change modifies the definitions of RTS and ESF response time to allow a substitution of an allocated response time for selected sensors in lieu of measuring the response time. The allocated time adequately represents the actual measured time for the associated sensors. The overall response time of each channel will continue to be measured using a series of sequential, overlapping or entire channel measurements to ensure the components actuated by each channel perform their accident mitigation function within the response time assumed in the safety analysis.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* September 21, 2001.

*Description of amendment request:* Entergy Operations, Inc. (Entergy, the licensee) is requesting approval of changes to the Waterford Steam Electric Station, Unit 3, Operating License and Technical Specifications associated with an increase in the licensed power level. The changes involve a proposed increase in the power level from 3,390 Megawatts thermal (MWt) to 3,441 MWt. These changes result from increased feedwater flow measurement accuracy to be achieved by utilizing high accuracy ultrasonic flow measurement instrumentation.

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

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

The comprehensive analytical efforts performed to support the proposed change included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by this change. All systems and components will function as designed, and the applicable performance requirements have been evaluated and found to be acceptable.

The primary loop components (reactor vessel, reactor internals, control element drive mechanisms, loop piping and supports, reactor coolant pumps, steam generators, and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. The Leak Before Break analysis conclusions remain valid, and thus the limiting break sizes determined in this analysis remain bounding. All of the NSSS will still perform the intended design functions during normal and accident conditions. The auxiliary systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS and Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The main steam safety valves (MSSVs) will provide adequate relief capacity to maintain the steam generator pressures within design limits. The atmospheric dump valves and steam bypass valves meet design sizing requirements at the uprated power level. The current Loss of Coolant Accident (LOCA) hydraulic forcing functions are still bounding for the proposed 1.5 percent increase in power.

Because the integrity of the plant will not be affected by operation at the uprated condition, it is concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the power calorimetric measurement allows the current safety analyses to be used, without change, to support operation at a core power of 3,441 megawatts thermal (MWt). As such, all Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to either bound operation at the 1.5 percent uprated condition, or new analyses were performed to verify all acceptance criteria continue to be met.

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

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

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed changes. The new installation of the LEFM [leading edge flow meter] CheckPlus system has been analyzed, and failures of this system will have no effect on any safety-related system or any systems, structures or components required for transient mitigation. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

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

3. Will operation of the facility in accordance with this proposed change

involve a significant reduction in a margin of safety?

Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Docket Nos. STN 50–454 and STN 50–455, Byron Station, Units 1 and 2, Ogle County, Illinois*

*Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois*

*Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois*

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

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

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

*Date of amendment request:* August 1, 2001.

*Description of amendment request:* The requested changes to the technical specifications (TSs) propose to revise requirements that have been superceded based on licensed operator training programs being accredited by the Institute for Nuclear Power Operations (INPO), promulgation of the revised 10 CFR part 55, Operators' Licenses, and adoption of a systems approach to training as required by 10 CFR 50.120, Training and qualification of nuclear power plant personnel. The same changes were requested by AmerGen Energy Company, LLC (AmerGen) for the Clinton Power Station, Oyster Creek, and Three Mile Island, Unit 1. The proposed no significant hazards consideration for those plants is published elsewhere in the **Federal Register** under AmerGen.

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

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

There will be no changes to the procedures by which the operators operate the plants. There will be no changes to the systems, structures, or components in the plants.

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

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

There will be no changes to the procedures by which the operators operate the plants. There will be no changes to the systems, structures, or components in the plants.

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

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

There will be no change in the plants' systems, structures, or components, nor in the way in which they will be operated as a result of the proposed changes. Therefore, the proposed changes will not involve a significant reduction in a margin of safety.

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

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

*NRC Section Chiefs:* Anthony J. Mendiola, James W. Clifford.

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

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

*Date of amendment request:* September 21, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated September 21, 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:* Mr. Edward J. Cullen, Vice President, General Counsel, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348.

*NRC Section Chief:* Anthony J. Mendiola.

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

*Date of amendment request:* June 28, 2001.

*Description of amendment request:* The proposed amendment would revise the technical specification (TS) 3.1.1.4 upper limit for the moderator temperature coefficient (MTC) from $0 \times 10^{-4}$ change in reactivity per degree Fahrenheit (“k/k/°F) to $+0.2 \times 10^{-4}$ “k/k/°F for power levels up to 70 percent of rated thermal power (RTP), and ramping linearly to $0 \times 10^{-4}$ “k/k/°F from 70 percent to 100 percent RTP. The proposed change is needed to address future core designs with higher energy requirements, associated with plant operation at higher capacity factors.

*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 change from a[n] MTC of $0 \times 10^{-4}$ “k/k/°F to a positive moderator temperature coefficient (PMTC) of $+0.2 \times 10^{-4}$ “k/k/°F does not introduce an initiator of any design basis accident or event. The proposed change does not adversely affect accident initiators or precursors nor alter the configuration of the facility or the manner in which the plant is maintained. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change to a PMTC does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change is consistent with the safety analysis assumptions and resultant consequences. Accident analyses affected by the proposed change have been reanalyzed and all applicable acceptance criteria have been met. Thus, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed amendment 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 change to a PMTC does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed), subsequently no new or different failure modes or limiting single failures are created. The plant will not be operated in a different manner due to the proposed change. All SSCs will continue to function as currently designed. Thus, the proposed change does not create any new or different accident scenarios.

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

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

No. The proposed change to a PMTC does not involve revisions to any safety limits or safety system settings that would adversely impact plant safety. The proposed amendment does not alter the functional capabilities assumed in a safety analysis for any SSCs important to the mitigation and control of design bases accident conditions within the facility.

All of the applicable acceptance criteria (i.e., preventing reactor coolant system [RCS] or main steam system overpressurization, maintaining the minimum departure from nucleate boiling ratio [DNBR], preventing core uncovery, preventing fuel temperatures from exceeding their limit, preventing clad damage, and limiting the number of fuel rods that enter a departure from nucleate boiling [DNB] condition) for each of the analyses affected by the proposed change continue to be met. The conclusions of the UFSAR remain valid. Thus, since the operating parameters and system performance will remain within design requirements and safety analysis assumptions, safety margin is maintained.

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

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

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

*NRC Section Chief:* Lakshminaras Raghavan, (Acting).

*GPU Nuclear Inc., Docket No. 50–320, Three Mile Island Nuclear Generating Station, Unit 2, Dauphin County, Pennsylvania*

*Date of amendment request:* June 21, 2001.

*Description of amendment request:* The proposed technical specifications change request (TSCR) No. 81 is to revise Three Mile Island Nuclear Generating Station, Unit 2 (TMI–2) Technical Specification (TS) Administrative Controls section that will provide consistency with the changes to the revised 50.59 rule of Title 10 of the *Code of Federal Regulations* (10 CFR) Regulations, as published in the **Federal Register** on October 4, 1999 (64 FR 53582).

*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 reflect the revised 50.59 rule, issued as a Final Rule in October 4, 1999, and do not impact the operation of any system or component assumed in any accident analysis. The proposed change does not change the requirement to perform a 50.59 review when required by the Technical Specification Administrative Controls. Based on the administrative nature of this change there will be no direct impact on the radiological source term. Therefore, these changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

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

The proposed change does not impact or have a direct effect on any safety analysis assumptions. The proposed change is administrative in nature and makes the TS Administrative Control language conform to the Final Rule, dated October 4, 1999, related to the 10 CFR 50.59 rule. Changes to the facility that result in meeting the criteria of 10 CFR 50.59 will still require NRC approval pursuant to 10 CFR 50.59.

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

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

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

*NRC Section Chief:* Robert A. Gramm.

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

*Date of amendment request:* October 5, 2001.

*Description of amendment request:* The licensee proposed to amend the Technical Specifications (TSs) to change the licensing basis requirement for establishing containment hydrogen monitoring ''within 30 minutes'' to ''within 3 hours'' of initiating emergency core cooling following a loss-of-coolant accident (LOCA).

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

1. The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Updated Safety Analysis Report (USAR) Chapter 15 accident analyses do not require or take credit for hydrogen monitoring to be established shortly after a loss of coolant accident (LOCA). Post-LOCA hydrogen production occurs over a long period of time, and an extension from 30 minutes to 3 hours for establishing hydrogen monitoring will have a positive impact on the ability of the operators to concentrate on their more immediate actions while having no negative impact on containment integrity or the long-term assessment efforts. Therefore, the proposed license amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Control room operators use the containment hydrogen monitors following a LOCA to establish hydrogen control measures should it become necessary. The proposed license amendment would not eliminate the requirement to establish hydrogen monitoring, but would allow it to be delayed until those actions required to mitigate the accident and verify proper operation of essential safety equipment have been completed. The proposed extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment will not create a new or different kind of accident from any accident previously evaluated.

3. The operation of Nine Mile Point Unit 2 in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

The need to establish hydrogen control measures will not be present within the first 3 hours following a LOCA since there will not be significant hydrogen accumulation. By extending the time allowed to establish containment hydrogen monitoring, the operators can remain focused on the actions necessary to mitigate the accident before directing their attention to hydrogen control measures and other long-term actions. The proposed extension maintains the requirement to establish hydrogen monitoring well before calculated conditions inside the containment indicate any need to initiate hydrogen control measures. Therefore, the proposed license amendment will not involve a significant reduction in a margin of safety.

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

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

*NRC Section Chief:* L. Raghavan, Acting.

*Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant (DCPP), Units 1 and 2, San Luis Obispo County, California*

*Date of amendment requests:* September 13, 2001.

*Description of amendment requests:* The proposed license amendments would revise Technical Specification (TS) 3.7.16, ''Spent Fuel Pool Boron Concentration,'' TS 3.7.17, ''Spent Fuel Assembly Storage—Region 1/Region 2,'' and TS 4.3, ''Fuel Storage'' for DCPP Units 1 and 2, to allow the use of credit for soluble boron in the spent fuel pool criticality analysis.

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

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

*Response:* No.

There is no increase in the probability of a fuel assembly drop accident in the spent fuel pool (SFP) when considering the presence of soluble boron in the SFP water for criticality control. The handling of the fuel assemblies does not change as a result of crediting soluble boron in the SFP.

There is no increase in the probability of the accidental misloading of a fuel assembly into the SFP racks when considering the presence of soluble boron in the SFP water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification (TS) SFP storage configuration limitations.

There is no increase in the consequences of an accidental drop or accidental misloading of a fuel assembly into the SFP racks because the criticality analysis demonstrates that the pool will remain subcritical following either event even if the pool contains a boron concentration less than that currently specified in the TS. The current TS limitation will ensure that an adequate SFP boron concentration will be maintained.

There is no increase in the probability of the loss of normal cooling to the SFP water considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

There is no increase in the consequences of a loss of normal SFP cooling because the 2,000 ppm boron concentration required by TS provides significant negative reactivity to provide subcritical margin such that the SFP $k_{eff}$ is maintained less than or equal to 0.95 up to boiling (212°F).

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

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

Response: No.

Spent fuel handling accidents are not new or different types of accidents; they have been analyzed in Section 15.5.22 of the Updated Final Safety Analysis Report (UFSAR).

Criticality accidents in the SFP are not new or different types of accidents; they have been analyzed in the UFSAR and in the Criticality Analysis reports associated with the specific license amendments for fuel enrichments up to 5.0 weight percent U–235.

Because soluble boron has always been required in the SFP water, and is currently required by TS, credit for soluble boron will have no effect on normal pool operation and maintenance. Crediting soluble boron in the SFP criticality analysis will only result in increased sampling to verify the boron concentration. This increased sampling frequency will not create the possibility of a new or different kind of accident.

The SFP dilution analysis demonstrates that a dilution which could increase the rack $k_{eff}$ to greater than 0.95 is not a credible event. Therefore, crediting soluble boron in the SFP criticality analysis will not result in the possibility of a new kind of accident.

Revised specifications continue to specify the requirements for SFP storage configurations. The only significant changes relate to the criteria for determining the storage configuration. Because the proposed SFP storage configuration limitations will be similar to those currently contained in the TS, the new limitations will not have any significant effect on normal SFP operations and maintenance and will not create the possibility of a new or different kind of accident. A SFP loading verification will continue to be performed to ensure that the SFP loading configuration meets the specified requirements.

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

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

Response: No.

The TS changes proposed by this license amendment request and the resulting spent fuel storage limitations will provide an adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality analysis performed for the Diablo Canyon Units 1 and 2 SFPs that includes technically supported margins.

While the criticality analysis utilized credit for soluble boron, storage configurations have been defined to ensure that the spent fuel rack $k_{eff}$ will be less than 1.0 with no soluble boron with a 95 percent probability at a 95 percent confidence level. Soluble boron credit is used to offset uncertainties, tolerances and off-normal conditions, and to provide subcritical margin such that the SFP $k_{eff}$ is maintained less than or equal to 0.95. Since $k_{eff}$ is less than or equal to 0.95, the current margin of safety is maintained.

A substantial reduction in the SFP soluble boron concentration that could lead to exceeding a $k_{eff}$ of 0.95 has been evaluated and shown not to be credible.

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

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

*Attorney for licensee:* Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

*NRC Section Chief:* Stephen Dembek.

*Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant (DCPP), Unit Nos. 1 and 2, San Luis Obispo County, California*

*Date of amendment requests:* September 13, 2001.

*Description of amendment requests:* The proposed license amendments would modify Technical Specification (TS) 5.5.9, ''Steam Generator Tube Surveillance Program,'' to allow extension of steam generator tube W star alternate repair criteria for DCPP Units 1 and 2, from Cycles 10 and 11 to Cycles 12 and 13.

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

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

Of the various accidents previously evaluated, the extension of the steam generator (SG) tube W star (W*) alternate repair criteria (ARC) through Cycles 12 and 13 only affects the steam generator tube rupture (SGTR) accident evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this evaluation.

For the SGTR accident, the required structural margins of the SG tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the Westinghouse explosive tube expansion (WEXTEX) region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, ''Bases for Plugging Degraded PWR Steam Generator Tubes,'' margins against burst are maintained for both normal and postulated accident conditions.

WCAP–14797, Revision 1, defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). The W* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the WEXTEX expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. The proposed changes do not affect other systems, structures, components, or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident.

The consequences of an SGTR accident are affected by the primary-to-secondary leakage flow during the accident. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the WEXTEX expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) in the W* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. No leakage has been observed in any in-situ test of W* indications identified to date. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage, that is, the combined leakage for all such tubes, plus the combined leakage developed by any other ARC, are maintained below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR 100 guideline values.

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

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

The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon continued implementation of the W* ARC.

Axial indications left in service shall have the upper crack tip below the top of the tubesheet (TTS) by at least the value of the nondestructive examination (NDE) uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the tubesheet secondary face, thereby minimizing the potential for free span cracking and demonstrating that an acceptable level of risk is maintained for tubes returned to service under W* ARC. This repair criteria is in addition to ensuring that the upper crack tip is located below the bottom of the WEXTEX transition by at least the NDE measurement uncertainty. Condition monitoring will verify that all tubes returned to service under W* ARC remain below the TTS, including an allowance for NDE uncertainty.

These changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

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

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

The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. RG 1.121 is used as the basis in the development of the W* ARC for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube-burst that are consistent with the requirements of Section III of the ASME Code.

For primarily axially oriented cracking located within the tubesheet, tube-burst is precluded due to the presence of the tubesheet. WCAP–14797, Revision 1, defines a length, W*, of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Application of the W* ARC will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the W* ARC.

Plugging of the SG tubes reduces the reactor coolant flow margin for core cooling. Continued implementation of W* ARC will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report Update or Bases of the plant Technical Specifications.

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

*Attorney for licensee:* Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

*NRC Section Chief:* Stephen Dembek.

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

*Date of amendment request:* August 17, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated August 17, 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 requests, the requested changes do not involve a significant hazards consideration.

The NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

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

*NRC Section Chief:* James W. Clifford.

*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:* October 1, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated October 1, 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:* Thomas G. Eppink, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218.

*NRC Section Chief:* Richard J. Laufer, Acting.

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

*Date of amendment request:* May 3, 2001.

*Description of amendment request:* The proposed amendments would relocate cycle-specific reactor coolant system Technical Specifications parameters limits to the Core Operating Limits Report. Also, a reference to the Refueling Operations Boron Concentration is added to TS 5.6.5 to correct an omission.

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

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

The proposed amendment is a programmatic and administrative change that does not physically alter plant systems, nor does it impact the performance of their functions. No new equipment is added nor is installed equipment being changed or operated in a different manner. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve an increase in the probability or consequences of any accident previously evaluated.

The cycle-specific limits in the Core Operating Limits Report (COLR) will continue to be controlled by the Farley Nuclear Plant (FNP) programs and procedures. Each accident analysis addressed in the Final Safety Analysis Report (FSAR) will be examined with respect to changes in the cycle dependent parameters, which are obtained from the use of Nuclear Regulatory Commission (NRC) approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be conducted per the requirements of 10 CFR 50.59, will ensure that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated. The safety limits imposed in Technical Specification (TS) 2.1 are consistent with the values stated in the FNP FSAR.

This change does not involve an increase in the probability or consequences of any accident previously evaluated.

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

Relocation of the cycle-specific parameters has no influence or impact on, nor does it contribute in any way to the probability or consequences of an accident. No plant equipment, function or plant operation will be altered as a result of this proposed change. The cycle-specific parameters are calculated using the NRC approved methods and submitted to the NRC to allow the staff to continue to trend the values of these limits. The TS will continue to require operation within the core operating limits and appropriate actions will be required if these limits are exceeded. The safety limits are maintained in the COLR and appropriate actions will be required if these limits are exceeded. In addition, the minimum limit for Reactor Coolant System flow will be retained in the TS. The safety limits imposed in TS 2.1 are consistent with the values stated in the FNP FSAR.

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

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

The margin of safety is not affected by the removal of cycle-specific core operating limits from the TS. The margin of safety presently provided by current TS limits remains unchanged. Appropriate measures exist to control the values of these cycle-specific limits. The proposed amendment continues to require operation within the core limits as obtained from NRC approved reload design methodologies and the actions to be taken if a limit is exceeded remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload will involve a 10 CFR 50.59 evaluation to assure that operation of the unit within the cycle-specific limits will not involve a significant reduction in the margin of safety.

The proposed changes to relocate cycle specific parameter limits to the COLRs will not affect plant design or system operating parameters, there is no detrimental impact on any equipment design parameters, and the plant will continue to operate within prescribed limits. The safety limits imposed in TS 2.1 are consistent with the values stated in the FNP FSAR.

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

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

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

*NRC Section Chief:* Richard J. Laufer, Acting.

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

*Date of amendment request:* June 5, 2001.

*Description of amendment request:* The proposed amendments would modify Technical Specifications (TS) Surveillance Requirement 3.4.14.1 to clarify that the frequency does not apply to Reactor Coolant System Pressure Isolation Valves in the Residual Heat Removal System flow path. Also, related TS Bases and editorial changes are part of this TS change.

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

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

The proposed change to Surveillance Requirement (SR) 3.4.14.1 clarifies that the requirement to test Reactor Coolant System (RCS) Pressure Isolation Valves (PIVs) following valve actuation due to automatic or manual action or flow through the valve does not apply to PIVs in the Residual Heat Removal (RHR) flow path. This resolves a source of potential confusion and ensures that the testing requirements are implemented consistent with the historical licensing basis for Farley and the Improved Technical Specification conversion NRC Safety Evaluation Report. The valves will continue to be tested for back leakage every 18 months. The proposed change does not affect the consequences of a previously analyzed accident since the magnitude and duration of analyzed events are not impacted by this change. Thus, the consequences of a previously evaluated accident are unchanged.

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

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

The proposed change involves no change to the physical plant. It allows for a clarification to the testing requirements to ensure that the historical licensing basis for Farley is maintained. These valves are tested every 18 months to ensure that the back leakage is within acceptable limits. This testing will continue. These changes do not impact the function of the valves.

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 changes do not involve a significant reduction in a margin of safety.

The physical plant is unaffected by this change. The proposed change does not impact accident offsite dose, containment pressure or temperature, emergency core cooling system (ECCS) or reactor protection system (RPS) settings or any other parameter that could affect a margin of safety. The clarification of the testing requirements ensures that future testing is consistent with the historical licensing basis for Farley.

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

*NRC Section Chief:* Richard J. Laufer, Acting.

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

*Date of amendment request:* September 10, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated September 10, 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:* Mr. Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

*NRC Section Chief:* Richard J. Laufer, Acting.

*Virginia Electric and Power Company, Docket Nos. 50–280 and 50–281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia*

*Date of amendment request:* September 10, 2001.

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

The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 11, 2000 (65 FR 49271) on possible amendments to eliminate PASS, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the consolidated line item improvement process. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the **Federal Register** on October 31, 2000 (65 FR 65018). The licensee affirmed the applicability of the following NSHC determination in its application dated September 10, 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:* Donald P. Irwin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

*NRC Section Chief:* Richard J. Laufer, Acting.

**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.*

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

*Date of application for amendments:* April 1, 2001, as supplemented by letter dated July 26, 2001.

*Brief description of amendments:* The amendment revises Technical Specifications (TS) section 5.0, ''Administrative Controls,'' by (1) clarifying new diesel fuel oil limits for water and sediment, (2) revising guidance on changes to TS Bases consistent with changes to 10 CFR 50.59, (3) adding clarification to the requirements for the Safety Function Determination Program, (4) adding the CENTS computer code to the list of analytical methods used to determine core operating limits, and (5) revising the Core Operating Limits Report list of references to approved topical reports.

*Date of issuance:* October 15, 2001.

*Effective date:* October 15, 2001, and shall be implemented within 60 days of the date of issuance.

*Amendment Nos.:* Unit 1–137, Unit 2–137, Unit 3–137.

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

*Date of initial notice in **Federal Register:*** May 2, 2001 (66 FR 22022).

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* October 4, 2000, as supplemented March 8, 2001, March 27, April 26, May 14, May 18, June 4, June 11, June 26, June 29, July 3, July 16 (2 letters), July 17, August 17, and September 20, 2001.

*Brief description of amendment:* The license amendment revises the Harris Nuclear Plant Technical Specifications to support the replacement of the current Westinghouse Model D4 steam generators with Westinghouse Model Delta 75 replacement steam generators and revises the accident analyses to adopt the alternate source term (AST) methodology, using the guidance of Nuclear Regulatory Commission (NRC) Regulatory Guide 1.183.

*Date of issuance:* October 12, 2001.

*Effective date:* October 12, 2001.

*Amendment No.:* 107.

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

*Date of initial notice in **Federal Register:*** For the steam generator replacement amendment request, the initial notice is dated November 1, 2000 (65 FR 65338). The March 8, 2001, March 27, April 26, May 14, May 18, June 4, June 11, June 26, June 29, July 3, July 16 (2 letters), and September 20, 2001, supplements contained clarifying information only, and did not change the initial no significant hazards consideration determination, or expand the scope of the initial application. The initial notice for the adoption of the AST methodology, using the guidance of NRC Regulatory Guide 1.183, was published on August 8, 2001 (66 FR 41612). The August 17, 2001, supplement contained clarifying information only, and did not change the initial no significant hazards consideration determination, or expand the scope of the initial application.

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

No significant hazards consideration comments received: No.

*Entergy Nuclear Indian Point 2 and Entergy Nuclear Operations, Docket Nos. 50–003 and 50–247, Indian Point Nuclear Generating Unit Nos. 1 and 2, Westchester County, New York*

*Date of application for amendment:* July 13, 2001.

*Brief description of amendment:* The amendments revise Technical Specification (TS) 4.1.8, ''High Radiation Area,'' for Indian Point Unit 1 and TS 6.12, ''High Radiation Area,'' for Indian Point Unit 2 to delete the administrative requirements for the control of access to high radiation areas. The control of access to these areas is assured by the licensee's radiation protection programs that comply with 10 CFR 20.1601 by using the alternate methods in NRC Regulatory Guide 8.38, ''Control of Access to High and Very High Radiation Areas in Nuclear Power Plants,'' June 1993.

*Date of issuance:* October 10, 2001.

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

*Amendment No.:* 51 and 221.

*Facility Operating License Nos. DPR–5 and DPR–26:* Amendments revised the Technical Specifications.

*Date of initial notice in* ***Federal Register:*** September 5, 2001 (66 FR 46477).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2001.

No significant hazards consideration comments received: No.

*Entergy Nuclear Operations, Docket Nos. 50–003 and 50–247, Indian Point Nuclear Generating Unit Nos. 1 and 2, Westchester County, New York*

*Date of application for amendments:* December 12, 2000, as supplemented on April 12, April 16, May 24, June 6, and June 8, 2001.

*Brief description of amendments:* The conforming amendments reflected the transfer of the licenses, formerly held by Consolidated Edison Company of New York, Inc., to Entergy Nuclear Indian Point 2, LLC, as the owner of Indian Point 1 and 2, and to Entergy Nuclear Operations, Inc., as the entity authorized to maintain Indian Point 1 and operate Indian Point 2. The amendments were approved pursuant to Section 50.90 of Title 10 of the Code of Federal Regulations.

*Date of issuance:* September 6, 2001.

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

*Amendment Nos.:* 50 (Indian Point 1) and 220 (Indian Point 2).

*Facility Operating License Nos. DPR–05 and DPR–26:* Amendments revised the Licenses and Technical Specifications.

*Date of initial notice in* ***Federal Register:*** January 29, 2001 (66 FR 8122).

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

No significant hazards consideration comments received: Not applicable.

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

*Date of application for amendment:* June 12, 2001, as supplemented by letters dated July 31, September 19 and September 25, 2001.

*Brief description of amendment:* The amendment revised Technical Specification (TS) 3.8.1.1 to provide a one-time extension of the allowed outage time (AOT) for an inoperable emergency diesel generator (EDG) from three days to ten days. In addition, the amendment revised TS 3.4.4 to make the action associated with an inoperable emergency power supply to the pressurizer heaters consistent with the proposed EDG AOT.

*Date of issuance:* October 15, 2001.

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

*Amendment No.:* 234.

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

*Date of initial notice in* ***Federal Register:*** July 11, 2001 (66 FR 36341).

The July 31, September 19 and September 25, 2001, supplemental letters provided clarifying information and revised TSs that were 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 October 15, 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:* January 8, 2001.

*Brief description of amendment:* The change revises the lower limit of the allowable containment internal pressure in Technical Specification (TS) 3.6.1.4, ''Containment Systems—Internal Pressure,'' from 14.375 pounds per square inch, absolute (psia) to 14.275 psia.

*Date of issuance:* October 10, 2001.

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

*Amendment No.:* 174.

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

*Date of initial notice in* ***Federal Register:*** February 21, 2001 (66 FR 11058).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 10, 2001.

No significant hazards consideration comments received: No.

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

*Date of application for amendments:* October 10, 2000, as supplemented October 9, 2001.

*Brief description of amendments:* The amendments revised the licenses for Peach Bottom Units 2 and 3 to remove Atlantic City Electric Company as a licensee, in conjunction with the transfer of the minority ownership interests of Atlantic City Electric Company to the majority owners, Exelon Generation Company, LLC, and PSEG Nuclear LLC.

*Date of issuance:* October 18, 2001.

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

*Amendments Nos.:* 241 and 245.

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

*Date of initial notice in* ***Federal Register:*** November 27, 2000 (65 FR 70740).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 27, 2000.

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

*Date of application for amendment:* March 28, 2001, as supplemented July 19, and October 2, 2001.

*Brief description of amendment:* The amendment revised the Improved Technical Specifications 3.7.18, ''Control Complex Cooling System'' to allow a one-time increase in the completion time for restoring an inoperable Control Complex Cooling System train from 7 to 35 days.

*Date of issuance:* October 16, 2001.

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

*Amendment No.:* 200.

*Facility Operating License No. DPR–72:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** April 18, 2001 (66 FR 20006). The supplemental letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* February 21, 2000, as supplemented June 27, 2001.

*Brief description of amendment:* The amendments revise various administrative actions, requirements, and responsibilities contained in Improved Technical Specifications (ITS) 2.0, ''Safety Limits,'' and ITS 5.0, ''Administrative Controls,'' to reflect the recent CR–3 Nuclear Operations reorganization and the amended requirements of 10 CFR 50.72, 10 CFR 50.73 and 10 CFR 50.59.

*Date of issuance:* October 18, 2001.

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

*Amendment No.:* 201.

*Facility Operating License No. DPR–72:* Amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 21, 2001 (66 FR 15926). The supplemental 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 October 20, 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:* April 17, 2001.

*Brief description of amendments:* Minor changes and corrections to a Unit 1 license condition and to Technical Specifications of both Unit 1 and 2 to correct administrative errors, or to incorporate changes justified by previous submittals, or to correct logic errors, or to delete obsolete terminology and provide conforming changes to reflect the revisions to 10 CFR 50.59.

*Date of Issuance:* October 18, 2001.

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

*Amendment Nos.:* 177 and 119.

*Facility Operating License Nos. DPR–67 and NPF–16:* Amendments revised the Unit 1 Operating License and the Technical Specifications of both units.

*Date of initial notice in **Federal Register:*** May 30, 2001 (66 FR 29357).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 18, 2001.

No significant hazards consideration comments received: No.

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

*Date of amendment request:* February 28, 2001.

*Brief description of amendment:* The amendment revises the technical specifications to reflect changes in the standard by which the licensee will test charcoal used in engineered safety feature systems. The requested changes satisfy the requirements of NRC Generic Letter 99–02.

*Date of issuance:* October 16, 2001.

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

*Amendment No.:* 186.

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

*Date of initial notice in **Federal Register:*** June 12, 2001 (66 FR 31710).

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* February 27, 2001, as supplemented on September 6, 2001.

*Brief description of amendment:* The amendment revises surveillance requirements associated with Technical Specifications Section 3.3.8.2, ''Reactor Protection System (RPS) Electric Power Monitoring—Logic,'' and Section 3.3.8.3, ''Reactor Protection System (RPS) Electric Power Monitoring—Scram Solenoids.'' Specifically, the overvoltage allowable values and associated channel calibration frequency interval are changed.

*Date of issuance:* October 17, 2001.

*Effective date:* As of the date of issuance and shall be implemented by March 15, 2002.

*Amendment No.:* 99.

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

*Date of initial notice in **Federal Register:*** March 21, 2001 (66 FR 15928).

The staff's related evaluation of the amendment is contained in a Safety Evaluation dated October 17, 2001.

No significant hazards consideration comments received: No.

*PPL Susquehanna, LLC, Docket No. 50–388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania*

*Date of application for amendment:* November 16, 2000.

*Brief description of amendment:* This amendment deleted a note in TS Surveillance Requirement 3.6.1.1.1 which extended the leak rate testing surveillance interval on the 2S299A and 2S299B spectacle flange o-rings until the Unit 2 10th refueling outage or a prior Unit 2 outage requiring entry into Mode 4. The note, added in Amendment No. 160 to Facility Operating License No. NPF–22 which was issued on May 8, 2000, was necessitated because of a Notice of Enforcement Discretion documented in a letter dated April 11, 2000. This note is no longer required to be included in TS 3.6.1.1.1 because the surveillance test was conducted during the Unit 2 forced outage in August of 2000.

*Date of issuance:* October 9, 2001.

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

*Amendment No.:* 171.

*Facility Operating License No. NPF–22:* This amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 11, 2001 (66 FR 36343).

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* December 20, 1999, as supplemented on February 11, February 25, and October 10, 2000.

*Brief description of amendment:* The amendment revises the license to reflect changes related to the transfer of the license for the Hope Creek Generating Station, to the extent held by Atlantic City Electric Company to PSEG Nuclear LLC.

*Date of issuance:* October 18, 2001.

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

*Amendment No.:* 135.

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

*Date of initial notice in **Federal Register:*** February 18, 2000 (65 FR 8453).

The letters dated February 11, February 25, and October 10, 2000, 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 April 21, 2000.

No significant hazards consideration comments received: No.

*Public Service Electric & Gas Company, Docket Nos. 50–272 and 50–311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey*

*Date of application for amendments:* December 20, 1999, as supplemented February 11, February 25, and October 10, 2000.

*Brief description of amendments:* The amendments revised Facility Operating Licenses DPR–70 and DPR–75 to reflect changes related to the transfer of the license for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, to the extent held by the Atlantic City Electric Company, to PSEG Nuclear Limited Liability Company.

*Date of issuance:* October 18, 2001.

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

*Amendment Nos.:* 246 and 227.

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

*Date of initial notice in **Federal Register:*** February 18, 2000 (65 FR 8452). The February 11, February 25, and October 10, 2000, supplements did not expand the scope of the original application with respect to both the proposed transfer action and the proposed amendment action as initially noticed in the **Federal Register**. *No hearing requests or comments were received.* In addition, the submittals did not affect the applicability of the Commission's generic no significant hazards consideration determination set forth in 10 CFR 2.1315.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 21, 2000.

*Southern Nuclear Operating Company, Inc., Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket No. 50–366, Edwin I. Hatch Nuclear Plant, Unit 2, Appling County, Georgia*

*Date of application for amendment:* May 23, 2001.

*Brief description of amendments:* The amendment revises the Safety Limit Minimum Critical Power Ratio to reflect the results of a cycle-specific calculation that was performed using NRC-approved methodology.

*Date of issuance:* October 12, 2001.

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

*Amendment No.:* 167.

*Facility Operating License No. NPF–5:* Amendment revises the Technical Specifications.

*Date of initial notice in* **Federal Register:** June 12, 2001 (66 FR 31714).

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

No significant hazards consideration comments received: No.

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

*Date of amendment request:* April 25, 2001, as supplemented by letters dated July 31 and August 23, 2001.

*Brief description of amendments:* The amendments change the Technical Specifications (TS) to allow a one-time only change to TS 3.8.1, ''AC Sources—Operating,'' Action A.3, by extending the required Completion Time for restoration of an inoperable offsite circuit from 72 hours to 21 days.

*Date of issuance:* October 9, 2001.

*Effective date:* As of the date of issuance and shall be implemented no later than February 28, 2002.

*Amendment Nos.:* 88/88.

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

*Date of initial notice in* **Federal Register:** September 5, 2001 (66 FR 46482).

The supplemental letter dated August 23, 2001, provided clarifying information that did not change the Nuclear Regulatory Commission (the Commission) staff's proposed no significant hazards consideration determination.

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

No significant hazards consideration comments received: No.

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

*Date of application for amendment:* August 14, 2001, as supplemented on August 21, 2001.

*Brief description of amendment:* The proposed amendment would extend the allowed outage time (AOT) for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling systems from 7 days to 14 days. Requirements were added to immediately ensure the availability of alternate means of high pressure coolant makeup. Also, clarifying changes were made to Technical Specifications (TSs) 3.5.E.2 and 3.5.G.2 by reformatting the TSs to make the nomenclature consistent regarding HPCI and the Automatic Depressurization System as being systems, not subsystems.

*Date of Issuance:* October 18, 2001.

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

*Amendment No.:* 205.

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

*Date of initial notice in* **Federal Register:** September 18, 2001 (66 FR 48152).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 18, 2001.

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 22nd day of October 2001.

For the Nuclear Regulatory Commission.

**John A. Zwolinski,**

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

[FR Doc. 01–27261 Filed 10–30–01; 8:45 am]

**BILLING CODE 7590–01–P**

## SECURITIES AND EXCHANGE COMMISSION

**[Release No. 34–44972; File No. SR–Amex–2001–19]**

## Self-Regulatory Organizations; Notice of Filing of a Proposed Rule Change and Amendment Nos. 1, 2 and 3 by the American Stock Exchange LLC Relating to Its Performance Evaluation and Allocations Procedures

October 23, 2001.

Pursuant to section 19(b)(1) of the Securities Exchange Act of 1934 (''Act'') [1] and Rule 19b–4 thereunder,[2] notice is hereby given that on March 19, 2001, the American Stock Exchange LLC (''Amex'' or ''Exchange'') filed with the Securities and Exchange Commission (''SEC'' or ''Commission'') the proposed rule change as described in Items I, II, and III below, which Items have been prepared by the Exchange. On May 31, 2001, the Exchange submitted Amendment No. 1 to the proposed rule change.[3] On August 13, 2001, the Exchange submitted Amendment No. 2 to the proposed rule change.[4] On August 27, 2001, the Exchange submitted Amendment No. 3 to the proposed rule change.[5] The Commission is publishing this notice to solicit comments on the proposed rule change, as amended, from interested persons.

### I. Self-Regulatory Organization's Statement of the Terms of Substance of the Proposed Rule Change

The Exchange proposes to adopt Amex Rules 26 and 27 to codify the Exchange's performance evaluation and allocations procedures. The text of the proposed rule change is available at the Office of the Secretary, the Amex and the Commission.

### II. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change

In its filing with the Commission, the Exchange included statements concerning the purpose of, and basis for, the proposed rule change and discussed any comments it received on the proposed rule change. The text of these statements may be examined at the places specified in Item IV below. The Exchange has prepared summaries, set forth in Sections A, B, and C below, of the most significant aspects of such statements.

*A. Self-Regulatory Organization's Statement of the Purpose of, and Statutory Basis for, the Proposed Rule Change*

1. Purpose

The Board of Governors of the Exchange is generally responsible for the supervision of its members. With regards to (1) evaluating the performance of specialists, registered traders, and brokers, and (2) allocating securities to specialists, the Board has delegated its responsibilities to the Committee on Floor Member Performance (the ''Performance Committee'' or ''Committee'') and the Allocations Committee, respectively.[6]

Performance evaluation is the non-disciplinary process [7] by which the Exchange reviews Floor member conduct and takes remedial action where necessary to improve performance. The registration of specialists (''allocations'') is the process by which the Exchange matches appropriate specialists to particular securities.

The Exchange proposes to codify its performance evaluation and allocation procedures as Amex Rules 26 and 27 in order to make them readily available to members since these procedures currently are not available in one easily accessed location.

*Performance Evaluation (Rule 26)*

Paragraph (a) of proposed Rule 26 describes the composition of the Performance Committee. The proposed rule states that the Performance Committee consists of 16 persons drawn from a larger pool divided as equally as possible among specialists, registered traders, brokers and upstairs member firm representatives. Specialists, registered traders, and brokers are the three classes of market participants on the Exchange Floor. Upstairs member firm representatives, while not on the Floor, make extensive use of the Exchange's services and have another perspective on the operation of the market. A Floor Governor chairs meetings of the Performance Committee and only may vote to break a tie. A

[1] 15 U.S.C. 78s(b)(1).

[2] 17 CFR 240.19b–4.

[3] *See* Letter from Bill Floyd-Jones, Jr., Assistant General Counsel, Legal and Regulatory, Amex, to Katherine A. England, Assistant Director, Division Market Regulation (''Division''), Commission (May 31, 2001). Amendment No. 1 adds discussion to the purpose section of the proposal regarding the ability of the Performance Committee to take appropriate action should a member or member organization fail without a reasonable excuse to meet with the committee after receiving notice. In addition, Amendment No. 1 corrects structural and typographical errors that appeared in the proposed rule language.

[4] *See* Letter from Bill Floyd-Jones, Jr., Assistant General Counsel, Legal and Regulatory, Amex, to Katherine A. England, Assistant Director, Division, Commission (August 10, 2001). Amendment No. 2 adds a reference to the Special Allocations Committee in the proposal and proposed rule text; adds allocations procedures for structured products and Exchange Traded Funds; and makes technical changes to the proposed rule text.

[5] *See* Letter from Bill Floyd-Jones, Jr., Assistant General Counsel, Legal and Regulatory, Amex, to Katherine A. England, Assistant Director, Division, Commission (August 24, 2001). Amendment No. 3 clarifies the Performance and Allocations Committee review procedures.

[6] *See* Amex Rules 170 and 958, which establish standards for specialists and Registered Options Traders. *See also* Article II, Section 3 of the Exchange Constitution, which provides in relevant part:

The Board shall establish standards and requirements for the registration of specialists or odd-lot dealers in securities dealt in on the Exchange, and may grant to a committee or committees, the authority to (i) approve the registration of specialists or odd-lot dealers, (ii) revoke or suspend any such registration at any time, (iii) allocate to a registered specialist or odd-lot dealer any security dealt in on the Exchange, and (iv) revoke any such allocation, temporarily or permanently, at any time.

[7] *See In the Matter of the Application of Pacific Stock Exchange's Options Floor Post X–17,* Admin. Proc. File No. 3–7285, Securities Exchange Act Release No. 31666 (December 29, 1992), 51 SEC DOC 261. The Commission determined that performance evaluation processes fulfill a combination of business and regulatory interests at exchanges and are not disciplinary in nature. The Commission states in the *Post X–17* case:

We believe that the reallocation of a market maker's (or a specialist's) security due to poor performance is neither an action responding to a violation of an exchange rule nor an action where a sanction is sought or intended. Instead, we believe that performance-based security reallocations are instituted by exchanges to improve market maker performance and to ensure quality of markets. Accordingly, in approving rules for performance-based reallocations, we historically have taken the position that the reallocation of a specialist's or a market maker's security due to inadequate performance does not constitute a disciplinary sanction.

We believe that an SRO's need to evaluate market maker and specialist performance arises from both business and regulatory interests in ensuring adequate market making performance by its market makers and specialists that are distinct from the SRO's enforcement interests in disciplining members who violate SRO or Commission Rules. An exchange has an obligation to ensure that its market makers or specialists are contributing to the maintenance of fair and orderly markets in its securities. In addition, an exchange has an interest in ensuring that the services provided by its members attract buyers and sellers to the exchange. To effectuate both purposes, an SRO needs to be able to evaluate the performance of its market makers or specialists and transfer securities from poor performing units to the better performing units. This type of action is very different from a disciplinary proceeding where a sanction is meted out to remedy a specific rule violation. (Footnotes omitted.)