rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows: *Wednesday, October 3, 2001—10:00 a.m. until the conclusion of business.*

The Subcommittee will discuss proposed ACRS activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff person named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting the cognizant ACRS staff person, Howard J. Larson (telephone: 301/415–6805) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any changes in schedule, etc., that may have occurred.

Dated: September 12, 2001.

Howard J. Larson,

Special Assistant.

[FR Doc. 01–23333 Filed 9–18–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

Subcommittee Meeting on Thermal-Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal-Hydraulic Phenomena will hold a meeting on September 26–27, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

Portions of the meeting may be closed to public attendance to discuss General Electric (GE) Nuclear Energy proprietary information per 5 U.S.C. 552b(c)(4). The agenda for the subject meeting shall be as follows: *Wednesday, September 26, 2001—1:00 p.m. until the conclusion of business, Thursday, September 27, 2001—8:30 a.m. until the conclusion of business.*

The Subcommittee will review the license amendment request of the Nuclear Management Company, LLC, for a core power uprate for the Duane Arnold Energy Center. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman. Written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, Nuclear Management Company, LLC, GE Nuclear Energy, and other interested persons regarding this review.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301–415–8065) between 7:30 a.m. and 4:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: September 12, 2001.

Howard J. Larson,

Special Assistant.

[FR Doc. 01–23334 Filed 9–18–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 27, 2001 through September 7, 2001. The last biweekly notice was published on September 5, 2001 (66 FR 46473).

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

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

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

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

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

By October 19, 2001, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available records will be accessible and electronically from the ADAMS Public Library component on the NRC Web site, *http://www.nrc.gov* (the Electronic Reading Room). If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

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

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

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

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

Date of amendment request: July 23, 2001.

Description of amendment request: As a follow-up response to a commitment identified in the Nuclear Regulatory Commission (NRC) staff letter dated December 22, 2000, ''Completion of Licensing Action for Generic Letter (GL) 96–06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,'' Entergy Operations Inc., (Entergy, the licensee) has proposed to revise their Waterford Steam Electric Station, Unit 3 (Waterford 3) Final Safety Analysis Report (FSAR) to resolve the ten containment penetrations susceptible to thermally induces overpressurization through an evaluation, detailed analysis, or installation of physical modifications prior to startup from the spring 2002 refueling outage. Entergy determined a change to Waterford 3's license basis, through procedural controls, risk analysis, and engineering analysis, for seven penetrations, as discussed in this license basis change request. Permanent resolution to the GL 96–06 issues for the remaining three penetrations could be satisfied through the installation of physical modifications.

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 proposed FSAR change reflects the use of administrative procedural controls to ensure these seven containment penetrations (two 4-inch diameter Steam Generator Blowdown penetrations and five-½ inch diameter Process Sampling penetrations) contain fluid at temperatures representative of Reactor Coolant, and the very low probability for overpressurization failure of containment penetrations during Mode 4 plant operation as a permanent solution to the GL 96–06 issue. The engineering analysis determined these seven containment penetrations met the acceptance criteria for allowed stresses contained in ASME [American Society of Mechanical Engineers] Section III Code, [Boiler and Pressure Vessel Code] Appendix F 1995. The result of the risk analysis is such that the very small change in LERF (Large Early Release Frequency], on the order of 1×10^{-9} per reactor year, remained well below the 1×10^{-7} ΔLERF guideline for a small change given in Regulatory Guide 1.174. The negligible reduction in LERF that would be achieved by adding thermal relief valve overpressure protection is not risk significant and is too small to justify the addition of the relief valves.

With respect to the probability or the consequences of an accident previously evaluated in the FSAR, the proposed deviation to the existing ASME Section III Code, Class 2 design provisions and operating requirements for the seven containment penetrations would not significantly increase the probability of an accident since the administrative procedural controls are being provided to: (1) minimize penetration heat-up and over-pressurization during a small window of vulnerability, approximately 1% per year of Mode 4 plant operation; and (2) minimize process fluid cooldown during normal plant operation by closing the containment isolation valves for the five sample penetrations when process fluid samples are obtained and the laboratory sample valves downstream of the CIV [containment isolation valves] are closed or flow through the penetration is stopped. Also the results of engineering analyses showed that the containment penetrations may exceed ASME Section III, Subsection NC 3500 Code required yield stresses and experience plastic deformation, but would not catastrophically fail; therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity.

On this basis, the proposed changes are not considered to constitute a significant increase in the probability or consequences of an accident due to:

- Administrative controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability
- The seven containment penetrations retaining their ability to perform their safety function and maintaining containment integrity in accordance with engineering analyses performed that met acceptance criteria for allowed stresses contained in ASME Section III Code, Appendix F 1995, and
- The low risk significance of overpressurization failure of the seven containment penetrations during a DBA [Design Basis Accident] while the plant is in Mode 4.

The proposed changes will not significantly affect the results of any accident previously evaluated. The accident mitigation features of the plant are not significantly affected by these proposed changes. The proposed changes do not add or modify any existing equipment.

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 change proposes a deviation to the existing ASME, Section III, Class 2 license basis requirements for portions of the Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System that penetrate the containment as a permanent solution to the GL 96–06 issues. This change involves recognition of the acceptability of administrative procedural controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability, approximately 1% per year for Mode 4 plant operation. Added assurance is provided through the engineering analysis performed on these penetrations that determined allowable stresses did not exceed the ASME Section III Code, Appendix F 1995 pipe stress values. Therefore, the change would not contribute to the possibility of, or be the initiator for any new or different kind of accident.

The proposed change does not alter the configuration of the plant. There has been no physical change to plant systems, structures, or components.

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 does not involve a significant reduction in margin of safety. The existing licensing basis for Waterford 3, with respect to the ASME Section III, Subsection NC–3621.2 provisions for portions of the Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System that penetrate the containment, is to ensure piping that has the potential to experience pressurization due to trapped fluid expansion shall be designed to withstand the increased pressure or have provisions for relieving the excess pressure piping. With the acceptance of this proposed deviation to the license basis, it will be recognized that the seven containment penetrations have administrative procedural controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability, approximately 1% per year for Mode 4 plant operation. Added assurance is also provided through the engineering analysis performed on these penetrations that determined stresses did not exceed the ASME Section III Code, Appendix F 1995 pipe stress values and predicted the penetration piping would experience plastic deformation, but would not catastrophically fail. Therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity. This deviation to license basis requirements for these seven containment penetrations is not considered to constitute a significant decrease in the margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 13, 2001.

Description of amendment request: The proposed amendments delete requirements from the Technical Specifications (TSs) 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 August 13, 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 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: Mary O'Reilly, FirstEnergy Nuclear Operating Company, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Timothy G. Colburn, Acting.

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 amendment request: August 22, 2001.

Description of amendment request: Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR–67 for St. Lucie Unit I and NPF–16 for St. Lucie Unit 2 by revising Technical Specifications (TS) relating to positive reactivity additions while in shutdown modes. The proposed changes clarify TS involving positive reactivity additions to the shutdown reactor, and would allow small, controlled, safe insertions of positive reactivity while in shutdown modes. The proposed changes conform closely to an NRC approved generic change for Standard Technical Specifications, known as TSTF–286 Rev. 2, which revises most actions requiring ''Suspend operations involving positive reactivity additions'' to allow minimum reactivity additions due to temperature fluctuations or operations, which are necessary to maintain fluid inventory within the required shutdown margin or refueling boron concentration, as 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) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes revise actions that either require suspension of operations involving positive reactivity additions or preclude reduction in boron concentration less than the reactor coolant system (RCS). Reactivity excursions are analyzed events. The proposed changes limit positive reactivity additions into the RCS such that the required shutdown margin (SDM) or refueling boron concentration continue to be met. Reactivity changes performed during shutdown modes are currently governed by strict administrative controls. Although the proposed changes will allow procedural flexibility with regards to RCS temperature and boron concentration, these operations will still be under administrative control. The changes proposed by these amendments are within the scope and assumptions of the existing analyses. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed TS revisions relate to positive reactivity additions while in shutdown modes of operation. Reactivity excursions are analyzed events. The operational flexibility allowed in these proposed license amendments will be performed under strict administrative controls in order to limit the potential for excess positive reactivity addition. Although the existing procedural controls will need modification, no new or different operational failure modes would be introduced by these changes.

Additionally, implementation of these proposed changes do not require any physical plant modifications, so no new or different hardware related failure modes are introduced. The changes proposed by these amendments are within the scope and assumptions of the existing analyses. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes conform closely to the industry and NRC approved TSTF–286, Rev. 2 and relate to small, controlled, safe insertions of positive reactivity additions while in shutdown modes. These changes revise actions that either require suspension of operations involving positive reactivity additions, or prohibit RCS boron concentration reduction. The proposed changes provide operational flexibility while controlling positive reactivity additions in order to preserve the required SDM or refueling boron concentration. The proposed changes to provide for continued safe reactor operations, while also limiting any potential for excess positive reactivity addition. Therefore, operation of the facility in accordance with the proposed amendments 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

Florida Power and Light Company, Docket No. 50–389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of amendment request: June 22, 2001, as supplemented August 24, 2001.

Description of amendment request: The proposed amendment would revise the St. Lucie Unit 2 Technical Specification (TS) 3.9.4, Containment Penetrations. TS 3.9.4.a. requires that the containment equipment door be closed during core alterations or movement of irradiated fuel within containment. TS 3.9.4.b. requires a minimum of one door in each airlock to be closed during core alterations or movement of irradiated fuel within containment. The proposed change to TS 3.9.4.a. would allow the containment equipment door to be open during core alterations and movement of irradiated fuel in containment provided: (a) The equipment door is capable of being closed with four bolts within 30 minutes, (b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and (c) a designated crew is available at the equipment door to close the door. The capability to close the containment equipment door includes the requirements that the door is capable of being closed and that any cables or hoses across the equipment door have quick-disconnects to ensure the door is capable of being closed in a timely manner. The proposed change to TS 3.9.4.b would allow both doors of each containment airlock to be open during core alterations and movement of irradiated fuel in containment provided: (a) At least one door of each open containment airlock is capable of being closed, (b) the plant is in MODE 6 with at least 23 feet of water above the reactor pressure vessel flange, and (c) a designated individual is available outside each open containment airlock to close the door. The capability to close the containment airlock door includes the requirement that the door is capable of being closed and that any cables or hoses across the airlock door have quick-disconnects to ensure the door is capable of being closed in a timely manner.

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

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

The proposed change to TS 3.9.4 would allow the containment equipment door and both doors of each containment airlock to be open during fuel movement or core alterations. Currently, the equipment door is closed with four (4) bolts and a single door on each containment airlock is closed during fuel movement or core alterations to prevent the escape of radioactive material in the event of an in-containment fuel handling accident. Neither the containment equipment door nor either of the containment airlock doors is an initiator of an accident. Whether the containment equipment door or both doors of the containment air locks are open or closed during fuel movement and core alterations has no affect on the probability of any accident previously evaluated. Allowing the containment equipment door and the containment airlock doors to be open during fuel movement or core alterations does not significantly increase the consequences from a fuel handling accident. The calculated offsite doses are well within the limits of 10 CFR part 100. In addition, the calculated doses are larger than the expected doses because the calculation does not incorporate the closing of the containment equipment door or the containment airlock doors after the containment is evacuated, which would be much less than the two hours assumed in the analysis. The proposed change would significantly reduce the dose to workers in containment in the event of a fuel handling accident by reducing the time required to evacuate the containment. The changes being proposed do not affect assumptions contained in other plant safety analyses or the physical design of the plant, nor do they affect other Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

The proposed change to Technical Specification 3.9.4, ''Containment Building Penetrations,'' affects a previously evaluated fuel handling accident. The new Fuel Handling Accident Analysis assumes that all of the iodine and noble gases that become airborne escape and reach the exclusion boundary and low population zone with no credit taken for filtration, the containment building barrier or for decay or deposition. Since the proposed change does not involve the addition or modification of equipment nor does it alter the design of plant systems and the revised analysis is consistent with the Fuel Handling Accident Analysis, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety as defined by 10 CFR part 100 has not been significantly reduced. The calculated dose is well within the limits given in 10 CFR part 100 or NUREG 0800. The proposed change does not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: February 28, 2001.

Description of amendment request: The proposed amendment to the Cooper Nuclear Station (CNS) Operating License DPR–46 would revise the design basis accidents (DBA) radiological assessment methodology for offsite and control room radiological doses, and the associated supporting Technical Specifications (TS).

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

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

The proposed revisions to the CNS DBA radiological assessment methodology for offsite and control room doses, and the associated supporting TS changes, do not involve initiators or precursors of accidents previously evaluated. Furthermore, these changes do not affect the design, function, or modes of operation of systems, structures, or components within the facility. Therefore, the proposed radiological assessment calculational methodology revisions and TS changes do not involve a significant increase in the probability of an accident previously evaluated in the Updated Safety Analysis Report (USAR).

The proposed revisions to the CNS DBA radiological assessment methodology for offsite and control room doses, and the associated supporting TS changes, do not affect the design, function or modes of operation of systems, structures or components in the facility. The calculation revisions utilize conservatively lower accident mitigation system filter efficiency assumptions and incorporate plant specific accident mitigation system operating parameter and design assumptions. Due to the changes in the calculational methodology and assumptions, and an increase in the postulated accident source term, the calculated radiological dose consequences of each DBA have changed and in some cases increased. In each case, however, the calculated radiological dose consequences are within the exclusion area boundary (EAB) and low population zone (LPZ) radiological dose acceptance criteria specified in 10 CFR part 100 and the control room dose acceptance criteria discussed in General Design Criterion (GDC) 19 of 10 CFR part 50, Appendix A. Therefore, the proposed revisions to the radiological assessment methodology, and associated TS changes, do not involve a significant increase in the consequences of an accident previously evaluated in the USAR.

2. Does not create the possibility for a new or different kind of accident from any accident previously evaluated.

The proposed revisions to the CNS DBA radiological assessment methodology for offsite and control room doses, and the associated supporting TS changes, do not affect the design, function or mode of operation of systems, structures or components in the facility such that new equipment failure modes are created. No new or different type of plant equipment is installed by the revised radiological assessment calculational methodology or changes to the TS. Neither the calculations nor the TS changes introduce changes to existing design parameters governing normal plant operation or new plant operating modes. No new types of accident initiators or precursors are created by the proposed revisions. Therefore, the proposed revisions to radiological assessment methodology and the proposed changes to the TS do not create the possibility of a new or different kind of accident previously evaluated in the USAR.

3. Does not create a significant reduction in the margin of safety.

The proposed revisions to the CNS DBA radiological assessment methodology for offsite and control room doses, and the associated supporting TS changes, do not affect the design, function or mode of operation of systems, structures or components in the facility. These proposed TS changes are consistent with the criteria of 10 CFR 50.36(c)(2)(ii) for TS content.

The proposed revisions will not result in any challenges to plant equipment, fuel integrity, or the reactor coolant system pressure boundary. Due to the changes in the calculational methodology and assumptions, and an increase in the postulated accident source term, the calculated radiological dose consequences of each design basis accident have changed and in some cases increased. In each case, however, the calculated radiological dose consequences are within the EAB and LPZ radiological dose acceptance criteria specified in 10 CFR part 100 and the control room dose acceptance criteria discussed in GDC 19 of 10 CFR part 50, Appendix A. Therefore, the proposed revisions to the radiological assessment methodology, and associated TS changes, do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 12, 2001.

Description of amendment request: The proposed amendment would change the Cooper Nuclear Station (CNS) Technical Specification (TS) 5.5.10.b.2 to replace the phrase, ''A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59'' with the phrase ''A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.''

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

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

The proposed change deletes the reference to unreviewed safety question as defined in 10 CFR 50.59. Deletion of the definition of unreviewed safety question was approved by the NRC with the revisions to 10 CFR 50.59. Consequently, the probability of an accident previously evaluated is not significantly increased. Changes to the TS Bases are still evaluated in accordance with 10 CFR 50.59. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce the margin of safety because it has no direct effect on any safety analyses assumptions. Changes to the TS Bases that result in meeting the criteria in revised 10 CFR 50.59 (c)(2) will still require NRC approval pursuant to 10 CFR 50.59. This change is administrative in nature as discussed by the NRC in FR (Volume 64, Number 191, Pages 53582–53617) dated October 4, 1999, docketing the change to 10 CFR 50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April, 12, 2001.

Description of amendment request: The amendment request would modify the Cooper Nuclear Station (CNS) Technical Specifications Surveillance Requirement (SR) 3.6.1.3.8 to relax the SR frequency by allowing a representative sample of Excess Flow Check Valves (EFCVs) to be tested every 18 months, such that each EFCV will be tested once every 10 years.

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

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

The current SR frequency requires each reactor instrumentation line EFCV to be tested every 18 months. The EFCVs at CNS are designed to close automatically in the event of a line break downstream of the valve. This proposed change allows a reduced number of EFCVs to be tested every 18 months. Industry operating experience, documented in BWR [Boiling Water Reactor] Owners' Group Topical Report NEDO–32977-A [''Excess Flow Check Valve Testing Relaxation,'' dated June 2000], concludes that a change in surveillance test frequency has a minimal impact on the reliability for these valves. A failure of an EFCV to isolate cannot initiate previously evaluated accidents. Furthermore, neither the EFCV actuation test, nor the frequency of testing is considered an initiator of any analyzed event. Therefore, there is no increase in the probability of occurrence of an accident as a result of this proposed change.

The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. The installed restricting orifice on each associated instrument line provides assurance that any instrument line break will limit offsite doses to substantially below 10 CFR part 100 values. Neither the EFCV actuation test, nor the frequency of testing is an analysis assumption. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the radiological consequences of an accident previously evaluated as a result of this change.

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

This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in [BWR Owners' Group Topical Report NEDO–32977–A] provides supporting evidence that the reduced testing frequency will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line pipe break. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

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

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. EFCV design, operation, and flow actuation criteria remain unaffected by this change. Restricting orifices for each associated instrument line remains available to mitigate an instrument line break. The proposed change, which impacts the frequency of testing EFCVs is acceptable because the tests continue to require appropriate confirmation of the assumed function of the system (and thereby assure continued operability), and has been shown to reflect an acceptable frequency for detecting failures. There is no detrimental impact on any other equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: August 9, 2001.

Description of amendment request: The amendment would change the Seabrook Station Technical Specifications (TSs) Index, TS 3/4.9.3 (''Decay Time''), TS 3/4.9.4 (''Containment Building Penetrations''), and TS 3/4.9.9 (''Containment Purge And Exhaust Isolation System''). The amendment would also change Bases 3/4.9.3, Bases 3/4.9.4, and Bases 3/4.9.9 for consistency with the proposed TS changes. These changes are consistent with the improved Standard Technical Specifications (STS) for Westinghouse plants.

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

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

The proposed changes to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not adversely affect accident initiators or precursors nor do they adversely alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. In addition, the proposed changes do not adversely affect the manner in which the plant responds in normal operation, transient or accident conditions nor do they change any of the procedures related to operation of the plant. Though a portion of the proposed change to TS 3/4.9.4 appears to be a relaxation to the current licensing basis, North Atlantic has incorporated administrative conservatism into TS 3/4.9.4 to assure the proposed changes, in conjunction with other TS required surveillance testing, do not alter or prevent the ability of structures, systems and components (SSCs), in particular the Containment Purge and Exhaust Isolation System, to perform its intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR).

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

Therefore, it is concluded that these proposed revisions to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

This proposed changes to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not adversely affect the operation nor do they change the design basis of any plant system or component during normal or accident conditions. The proposed changes do not include any physical changes to the plant. In addition, the proposed changes do not adversely affect the function or operation of plant equipment or introduce any new failure mechanisms such that the design basis is adversely affected. The current licensing basis allows penetration isolation by manual or automatic means. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes that creates the possibility of a new or different kind of accident.

The proposed changes do not modify the facility nor do they adversely affect the plant's response to normal, transient or accident conditions. The changes do not introduce a new mode of plant operation. While these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety. The plant's design and design basis are not revised and the current safety analyses remains in effect.

Thus, these proposed revisions to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not adversely affect the safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications nor is the plant design revised by the proposed changes. The current licensing basis allows penetration isolation by manual or automatic means.

Though a portion of the proposed change to TS 3/4.9.4 appears to be a relaxation to the current licensing basis, North Atlantic has incorporated administrative conservatism into TS 3/4.9.4 to ensure the proposed changes, in conjunction with other TS required surveillance testing, offset any potential minimal reduction in the margin of safety. North Atlantic believes that the proposed change to TS 3/4.9.4 is more conservative than that currently allowed in the improved STS, NUREG–1431, Revision 2.

Thus, it is concluded that these proposed revisions to TS Index, TS 3/4.9.3, TS 3/4.9.4, and TS 3/4.9.9 do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: August 15, 2001.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TSs) to (1) reflect the replacement of Monticello's licensed operator initial and requalification training programs with an accredited systems approach to training program and (2) relocate the existing TS requirements for procedures, records, and reviews to the operational quality assurance plan.

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

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

The proposed changes are administrative in nature and compliance with applicable regulatory requirements will continue to be maintained. The proposed changes do not involve any change to the configuration or alter existing system relationships. In addition, the proposed changes do not alter the conditions or assumptions in any of the previous accident analyses thus, the radiological consequences previously evaluated are not adversely affected by the proposed changes.

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

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

The proposed changes are administrative in nature and compliance with applicable regulatory requirements will continue to be maintained. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment. Accordingly, no new failure modes have been introduced for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Also, there will be no changes in types or increases in the amounts of any effluents released offsite.

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

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

The proposed changes are administrative in nature and do not involve any change in the methodology or method of operation of any plant equipment. The proposed changes do not involve any change to the configuration or alter existing system relationships. The appropriate controls to provide continued assurance of compliance to applicable regulatory requirements has been maintained.

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

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

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

NRC Section Chief: Claudia M. Craig.

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

Date of amendment request: August 31, 2001.

Description of amendment request: The proposed amendment would amend the licenses to change the required implementation date for previously issued Amendment No. 184 to Facility Operating License NPF–14 and Amendment No. 158 to Facility Operating License NPF–22. The proposed amendment would not alter any of the requirements of the Susquehanna Steam Electric Station (SSES) Unit 1 and 2 Technical Specifications (TSs). The previously issued amendments incorporate long-term power stability solution instrumentation into the SSES Unit 1 and 2 TSs. When implemented, these amendments will incorporate into the TSs the licensee's final response to GL 94–02, ''Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors.'' Specifically, these amendments will, in part, add TS requirements related to the operating power range monitoring (OPRM) system. The licensee stated that recently identified deficiencies in the OPRM trip setpoint methodology, as documented in a General Electric 10 CFR part 21 report issued on June 29, 2001, have adversely affected its ability to implement the subject amendments. Therefore, the licensee requested that the required implementation date for Amendment No. 184 to License No. NPF–14 and Amendment No. 158 to License No. NPF–22 be revised to become effective no later than November 1, 2003.

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 implementation date extension is administrative in nature and does not require any physical plant modifications, physically affect any plant systems or components, or entail changes in plant operation. The resulting consequences of transients and accidents will remain within the NRC approved criteria. Therefore, the proposed action does not involve an 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 amendment implementation date extension is administrative in nature and does not require any physical plant modifications, physically affect any plant systems or components, or entail changes in plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed amendment implementation date extension is administrative in nature and does not require any physical plant modifications, physically affect any plant systems or components, nor entail changes in plant operation. Since the proposed changes do not affect the physical plant or have any impact on plant operation, the proposed changes will not jeopardize or degrade the function or operation of any plant system or component. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Peter Tam, Acting.

Tennessee Valley Authority (TVA), Docket Nos. 50–260 and 50–296, Browns Ferry Nuclear Plant (BFN), Units 2 and 3, Limestone County, Alabama

Date of amendment request: August 17, 2001.

Description of amendment request: The proposed amendments would revise the reactor vessel pressure-temperature (P–T) limits depicted in Technical Specification Figure 3.4.9–1 for each unit. In addition, pursuant to 10 CFR 50.12, TVA is requesting an exemption from the requirements of 10 CFR part 50, Appendix G, to allow the use of American Society of Mechanical Engineers (ASME) Code Case N–640 as a basis for these revised curves. Code Case N–640, ''Alternative Requirement Fracture Toughness for Development of P–T Limit Curves for ASME Boiler and Pressure Vessel Code Section XI, Division 1,'' permits the use of the plane strain fracture toughness (K_{Ic}) curve instead of the crack arrest fracture toughness (K_{Ia}) curve for reactor pressure vessel materials in determining the P–T limits. The exemption request is being reviewed separately.

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

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

The proposed Units 2 and 3 change deals exclusively with the reactor vessel pressure-temperature (P–T) curves which define the permissible regions for operation and testing. Failure of the reactor vessel is not considered as a design basis accident. Through the design conservatisms used to calculate the P–T curves, reactor vessel failure has a low probability of occurrence and is not considered in the safety analyses. The proposed changes adjust the reference temperature for the limiting material to account for irradiation effects and provide the same level of protection as previously evaluated and approved. The adjusted reference temperature calculations were performed using the guidance contained in Regulatory Guide 1.99, Revision 2, and ASME Section XI Code Case N–640 to reflect use of the operating limits to 19.5 Effective Full Power Years (EFPY). These changes do not alter or prevent the operation of equipment required to mitigate any accident analyzed in the BFN Final Safety Analysis Report. Therefore, this change does not increase the probability or consequences of any previously evaluated accident.

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

The proposed change to the Units 2 and 3 reactor vessel P–T curves does not involve a modification to plant equipment. No new failure modes are introduced. There is no effect on the function of any plant system, and no new system interactions are introduced by this change. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed curves conform to the guidance contained in Regulatory Guide 1.99, Revision 2, and maintain the safety margins specified in 10 CFR 50, Appendix G. Therefore, the proposed amendment does not involve a reduction in a margin of safety.

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

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET l0H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: August 7, 2001 (TS–01–04).

Description of amendment request: The proposed amendment would add a new condition and associated actions to the Technical Specification Limiting Condition for Operation (LCO) 3.8.1, ''AC Sources Operating,'' to allow one Diesel Generator (DG) be out of service for 14 days.

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

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

The emergency DGs are designed as backup AC power sources in the event of loss of offsite power. The proposed AOT [allowed outage time] does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the DGs provide plant protection or which create new modes of plant operation. In addition, a Probabilistic Safety Analysis (PSA) evaluation concluded that the risk contribution of the AOT extension is non-risk significant. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce any new modes of plant operation or make physical changes to plant systems. Therefore, extension of the allowable AOT for DGs does not create the possibility of a new or different accident.

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

The DGs are designed as backup AC power sources in the event of loss of offsite power. The proposed AOT does not change the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the DGs provide plant protection or which create new modes of plant operation. In addition, a PSA evaluation concluded that the risk contribution of the AOT extension is non-risk significant. 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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 10H, Knoxville, Tennessee 37902.

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: August 20, 2001.

Description of amendment request: The proposed change to the Technical Specifications (TSs) would revise certain requirements associated with demonstrating the operability of alternate trains when redundant equipment is made or found to be inoperable. The TSs revised include: 4.4.B, 4.5.A.2, 4.5.A.3, 4.5.A.4, 4.5.B.2, 4.5.C.2, 4.5.C.3, 4.5.D.2, 4.5.D.3, 4.5.E.2, 4.5.F.2, 4.5.H.1, 4.7.B.3.c, 4.10.B.1, and 4.10.B.3.b.2. Some format and typographical errors are also being corrected.

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

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

Because changing surveillance test requirements does not change the probability of accident precursors, this proposed change does not affect the probability of an accident previously evaluated. Since other periodic and post-maintenance surveillance requirements ensure that the operability of systems and components is maintained, there is no significant increase in the consequences of accidents previously evaluated.

Furthermore, the removal of the additional surveillance testing from the Technical Specifications would result in a decrease in the probability of equipment failure because the excessive testing causes unnecessary wear on the safety-related equipment and unnecessary challenges to safety systems. Reduced testing may also eliminate the potential for human error associated with system alignments and misdirection of attention from monitoring and directing plant operations.

Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, do not increase the probability or consequences of accidents.

Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

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

Reduced surveillance testing does not create new or different kinds of accidents since modes of operation are unchanged and additional accident precursors are not introduced. System operability requirements and design bases remain the same, and reactor operations are unchanged. Since system and component testing only involves the assurance of operability, reduced testing does not introduce mechanisms that may contribute to the possibility of new or different kinds of accidents.

Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, do not create the possibility of new or different kinds of accidents.

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

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

The proposed change will not decrease operability requirements, nor reduce the equipment required during various plant conditions. An acceptable level of testing exists in other Technical Specification requirements to demonstrate system and component operability. There are no changes to system or component operability requirements; therefore, systems and components will be available to provide existing margins of safety. The same systems and components with the same performance levels assumed in safety analyses will still be available to mitigate consequences of postulated accidents.

Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, have no effect on margins of safety.

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

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

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

NRC Section Chief: James W. Clifford.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

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

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

Date of application for amendment: March 29, 2001, as supplemented by letters dated June 27, 2001, and July 24, 2001.

Brief description of amendment: The amendment revised the reactor coolant system heatup, cooldown, and inservice leak hydrostatic test limitations for the reactor coolant system to a maximum of 29 effective full power years in accordance with Title 10 of the Code of Federal Regulations, Part 50, Appendix G. These pressure-temperature (P-T) limits are contained in TMI Unit 1 Technical Specification (TS) 3.1.2. In addition, the amendment revised the low-temperature overpressure protection (LTOP) requirements in TSs 3.1.12 and 4.5.2 to reflect the revised P-T limits. These changes will allow operation of two reactor coolant pumps in a single loop during LTOP conditions.

Date of issuance: September 6, 2001.

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

Amendment No.: 234.

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

Date of initial notice in ***Federal Register:*** July 25, 2001 (66 FR 38758).

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

No significant hazards consideration comments received: No.

Calvert Cliffs Nuclear Power Plant, Inc., Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Docket No. 72–8, Calvert Cliffs Independent Spent Fuel Storage Installation, Calvert County, Maryland

Date of application for amendments: November 22, 1999, as supplemented by letters dated October 4 and November 10, 2000, and May 18, 2001.

Brief description of amendments: The amendments authorize revisions to the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report and Independent Spent Fuel Storage Installation Updated Safety Analysis Report to incorporate changes associated with the aircraft hazards analysis due to increased "random" military flights in the vicinity of these facilities. These changes constitute an unreviewed safety question as defined in 10 CFR 50.59 and 10 CFR 72.48.

Date of issuance: August 29, 2001.

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

Amendment Nos.: 246 and 221.

Renewed Facility Operating License Nos. DPR–53 and DPR–69 and Materials License No. SNM–2502: Amendments revised licenses.

Date of initial notice in ***Federal Register:*** *December 29, 1999 (64 FR 73085).*

The supplemental letters dated October 4 and November 10, 2000, and May 18, 2001, provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated August 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 11, 2000.

Brief description of amendment: The amendment revises the Technical Specifications (TSs) to incorporate editorial revisions, clarifications, and corrections. Specifically, the amendment: (1) Provides updated information and corrections to the TS cover page, table of contents, and list of figures, (2) revises TS 4.5.E, "Control Room Air Filtration System," to remove an incorrect system test description and provide consistent test values for system flow rate and filter efficiency, (3) revises TS 6.2.1.a, "Facility Management and

Technical Support,'' to reference the Quality Assurance Program Description as the location of the documentation rather than the Updated Final Safety Analysis Report, (4) revises TS 6.9.1.7, ''Monthly Operating Report,'' to change the recipient of the Monthly Operating Report, and (5) corrects the periodicity of the Radioactive Effluent Release Report from semi-annual to annual in TS 6.15, ''Offsite Dose Calculation Manual'' and TS 6.16, ''Major Changes to Radioactive Liquid, Gaseous and Solid Waste Systems.'' In addition, the amendment revises TS Figure 5.1–1B concerning the indicated vent location associated with Indian Point Unit 3 (IP3). The labels for the IP3 plant vent and the machine shop were reversed.

Date of issuance: August 29, 2001.

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

Amendment No.: 219.

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

Date of initial notice in ***Federal Register:*** *February 21, 2001 (66 FR 11057).*

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 23, 2001, as supplemented June 25, June 29, and July 19, 2001.

Brief description of amendment: The amendment revises pressure-temperature limit curves and cold overpressure protection limits.

Date of issuance: August 27, 2001.

Effective date: As of the date of issuance and shall be implemented within 90 days from the date of issuance. August 27, 2001.

Amendment No.: 197.

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

Date of initial notice in ***Federal Register:*** *July 11, 2001 (66 FR 36340).*

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

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 3, 2001.

Brief description of amendments: The amendments revised the PBAPS Units 2 and 3 Technical Specifications (TSs) to incorporate Technical Specification Task Force (TSTF) Item 258, Revision 4. TSTFs are changes to the improved standard TS that were initiated by the nuclear power industry and submitted to the NRC staff. TSTF–258, Revision 4, revises TS Section 5.0, Administrative Controls, to delete specific TS staffing requirements for licensed Reactor Operators (ROs) and Senior Reactor Operators (SROs), relocate the working hour limits to a plant procedure, clarify requirements for the Shift Technical Advisor position, add regulatory definitions for ROs and SROs, revise the Radioactive Effluent Controls Program to be consistent with the intent of 10 CFR Part 20, and revises radiological area control requirements for high radiation areas to be consistent with 10 CFR 20.1601(c).

Date of issuance: August 30, 2001.

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

Amendments Nos.: 240 and 243.

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

Date of initial notice in ***Federal Register:*** June 12, 2001 (66 FR 31708).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 30, 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: March 7, 2001, as supplemented April 25, June 20, and July 16, 2001.

Brief description of amendment: The amendment revised the Improved Technical Specifications (ITS) 5.6.2.20, ''Containment Leakage Rate Testing Program'' to allow a one-time interval increase for the Type A Integrated Leakage Rate Test for no more than 5 years.

Date of issuance: August 30, 2001.

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

Amendment No.: 197.

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

Date of initial notice in ***Federal Register:*** April 2, 2001 (66 FR 17967). 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 August 30, 2001.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket No. 50–316, Donald C. Cook Nuclear Plant, Unit 2, Berrien County, Michigan

Date of application for amendments: September 1, 2000.

Brief description of amendments: The amendment approves changes to the Updated Final Safety Analysis Report (UFSAR) regarding the modeling of the pressurizer heater operation and spray effectiveness as they relate to certain transients that are analyzed for pressurizer overfill. Specifically, the amendment approves a change to the moderator temperature coefficient currently in the UFSAR assumed as an initial condition for the loss of all nonemergency alternating current power and loss of normal feedwater transients.

Date of issuance: August 23, 2001.

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

Amendment No.: 237.

Facility Operating License No. DPR–74: Amendment revised the Updated Final Safety Analysis Report.

Date of initial notice in ***Federal Register:*** September 20, 2000 (65 FR 56953).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 23, 2001.

No significant hazards consideration comments received: No.

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

Date of application for amendment: January 18, 2001, as supplemented April 20, 2001.

Brief description of amendment: The amendment revises the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs) 3.10.m to increase the minimum reactor coolant flow from

85,500 gallons per minute (gpm) flow per loop to 93,000 gpm flow per loop.

Date of issuance: September 5, 2001.

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

Amendment No.: 157.

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

*Date of initial notice in **Federal Register:*** February 21, 2001 (66 FR 11062).

The April 20, 2001, supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original **Federal Register** notice.

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

No significant hazards consideration comments received: No.

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

Date of amendment request: June 18, 2001.

Brief description of amendment: The amendment deleted items 3 and 4 from Section 5.15, ''Post-Accident Radiological Sampling and Monitoring,'' of the Fort Calhoun Station, Unit No. 1 Technical Specifications, and thereby eliminates the requirements to have and maintain the post-accident sampling system (PASS).

Date of issuance: August 29, 2001.

Effective date: August 29, 2001, and shall be implemented within 120 days from the date of issuance.

Amendment No.: 200.

Facility Operating License No. DPR–40. The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 25, 2001 (66 FR 38765).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 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: April 11, 2001, as supplemented June 13, 2001.

Brief description of amendment: The amendment revises the Hope Creek Technical Specifications (TSs) to relax the frequency for testing of excess flow check valves (EFCVs). Specifically, TS surveillance requirement 4.6.3.4 has been changed to revise required testing of EFCVs from once per 18 months for all valves to a test of a representative sample each 18 months such that all valves are tested once in 10 years.

Date of issuance: August 28, 2001.

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

Amendment No.: 132.

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

*Date of initial notice in **Federal Register:*** May 30, 2001 (66 FR 29361).

The June 13, 2001, 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 August 28, 2001.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, et al., Docket Nos. 50–280 and 50–281, Surry Power Station, Units 1 and 2, Surry County, Virginia

Date of application for amendments: September 22, 2000.

Brief Description of amendments: These amendments revise the Facility Operating Licenses (FOLs) and the Technical Specifications (TS) to remove obsolete license conditions, make editorial changes in the FOLs, and implement associated changes to the TS and Bases.

Date of issuance: August 30, 2001.

Effective date: August 30, 2001.

Amendment Nos.: 227 and 227.

Facility Operating License Nos. DPR–32 and DPR–37: Amendments change the License and Technical Specifications.

*Date of initial notice in **Federal Register:*** November 1, 2000 (65 FR 65351).

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

No significant hazards consideration comments received: No.

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

Date of amendment request: March 22, 2001.

Brief description of amendment: The amendment (1) decreases the allowable values for Function 8, pressurizer pressure-low and pressurizer pressure-high, in Table 3.3.1–1, ''Reactor Trip System Instrumentation,'' and (2) increases the allowable value for Function 1.d, pressurizer pressure-low for safety injection, in Table 3.3.2–1, ''Engineered Safety Feature Actuation System Instrumentation.''

Date of issuance: August 30, 2001.

Effective date: August 30, 2001, and shall be implemented prior to entry into Mode 3 in the restart from refueling outage 12 scheduled for the Spring 2002.

Amendment No.: 140.

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

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

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

No significant hazards consideration comments received: No.

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

Dated at Rockville, Maryland, this 10th day of September, 2001.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 01–23209 Filed 9–18–01; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF PERSONNEL MANAGEMENT

Proposed Collection; Comment Request for Clearance of a Revised Information Collection: SF 3106 and SF 3106A

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: In accordance with the Paperwork Reduction Act of 1995 (Public Law 104–13, May 22, 1995), this notice announces that the Office of Personnel Management (OPM) intends to submit to the Office of Management and Budget (OMB) a request for clearance of a revised information collection. SF 3106, Application for Refund of Retirement Deductions/ Federal Employees Retirement System (FERS), is used by former Federal employees under FERS, to apply for a refund of retirement deductions