Dated: June 5, 2003.

D. L. Gamberoni,

Technical Coordinator, Office of the Secretary.

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NUCLEAR REGULATORY COMMISSION

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

I. Background

Pursuant to Pub. L. 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 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, May 16, 2003, through May 29, 2003. The last biweekly notice was published on May 27, 2003 (68 FR 28843).

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 Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By July 10, 2003, 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* 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 Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent 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 PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: January 29, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3.6.5.1, ''Drywell,'' Surveillance Requirement 3.6.5.1.3 to delay the performance of the next drywell bypass leakage test to no later than November 23, 2008. The proposed amendment would also revise TS 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to remove an exception which is no longer applicable and to reflect a one-time deferral of the primary containment Type A test to no later than November 23, 2008.

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?

Response: No.

The proposed changes will revise TS 3.6.5.1, ''Drywell,'' Surveillance Requirement SR 3.6.5.1.3 to delay the performance of the next drywell bypass leakage rate test (DBLRT) to no later than November 23, 2008. This request will also revise CPS [Clinton Power Station] TS 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than November 23, 2008. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. In addition, AmerGen is proposing to delete from TS 5.5.13 the expired exception that allowed deferral of the leakage rate testing of the primary containment penetration 1MC–042 until the seventh refueling outage.

The drywell houses the reactor pressure vessel, the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS), which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a Loss of Coolant Accident (LOCA) to the suppression pool, where it is condensed. Air forced from the drywell is released into the primary containment through the suppression pool. The suppression pool is a concentric open container of water with a stainless steel liner that is located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (SRV) discharges or from a LOCA.

The function of the Mark III containment is to isolate and contain fission products released from the RCS following a design basis LOCA and to confine the postulated release of radioactive material to within limits. The test interval associated with the drywell bypass leakage and Type A testing is not a precursor of any accident previously evaluated. Therefore, extending these test intervals on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of the drywell bypass leakage and Type A testing provides assurance that the CPS drywell and primary containment will not exceed allowable leakage rate values specified in the TS and will continue to perform its design function following an accident. The risk assessment of the proposed changes has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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

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

Response: No.

The proposed changes for a one-time extension of the drywell bypass leakage and Type A tests and deletion of an expired local leak rate test exception for CPS, will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed changes do not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

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

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

Response: No.

CPS is a General Electric BWR/6 plant with a Mark III containment system. The Mark III containment design is a single-barrier pressure containment and a multi-barrier fission containment system consisting of the drywell and primary containment. The drywell houses the reactor pressure vessel, the reactor coolant recirculating loops, and branch connections of the RCS, which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a LOCA to the suppression pool, where it is condensed. The suppression pool is an annular pool of demineralized water between the drywell and the outer primary containment boundary. This pool covers the horizontal vent openings in the drywell to maintain a water seal between the drywell interior and the remainder of the containment volume. The primary containment consists of a steel-lined, reinforced concrete vessel, which surrounds the RCS and provides an essentially leak-tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions. The primary containment is penetrated by access, piping and electrical penetrations.

The integrity of the drywell is periodically verified by performance of the DBLRT. This test ensures that the measured drywell bypass leakage is bounded by the safety analysis assumptions. The drywell integrity is further verified by a number of additional tests, including drywell airlock door seal leakage tests, overall drywell airlock leakage tests and periodical visual inspections of exposed accessible interior and exterior drywell surfaces. Additional confidence that significant degradation in the drywell leaktightness has not developed is provided by the periodic qualitative assessment of drywell performance.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.'' These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed changes for a one-time extension of the drywell bypass leakage and Type A tests and deletion of an expired local leak rate test exception for CPS, do not effect the method for drywell or containment testing or the test acceptance criteria.

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

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

Attorney for licensee: Edward J. Cullen, Deputy General Counsel Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendments request: April 25, 2003.

Description of amendments request: The amendments would revise Specification 5.3.1 in Section 5.3, ''Unit Staff Qualifications,'' of the Technical Specifications, and add a new Specification 5.3.2. Specification 5.3.1 states the qualifications of the unit staff. The revision would state there is an exception for operator license applicants and the new specification would provide the requirements for these applicants. Only the qualifications of operator license applicants are being changed. Because a new specification would be added, the existing Specification 5.3.2 would also be renumbered 5.3.3.

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

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

The proposed Technical Specification (TS) change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program. These changes conform to the current requirements of 10 CFR [Part] 55. The TS requirements for all other unit staff qualifications remain unchanged.

Although licensed operator qualifications and training may have an indirect impact on accidents [involving operator action] previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised 10 CFR [Part] 55 rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. Palo Verde's licensed operator training program is accredited by INPO [Institute of Nuclear Power Operations] and is based on a systems approach to training.

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

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

The proposed TS change is an administrative change to clarify the current requirements for licensed operator qualifications and licensed operator training program and to conform to the revised 10 CFR [Part] 55. The TS requirements for all other unit staff qualifications remain unchanged.

As noted above, although licensed operator qualifications and training may have an indirect impact on the possibility of a new or different kind of accident [involving operator action] from any accident previously evaluated, the NRC considered this impact during the rulemaking process, and by promulgation of the revised rule, concluded that this impact remains acceptable as long as the licensed operator training program is certified to be accredited and is based on a systems approach to training. [That is to say an accredited license operator training program that is based on a systems approach to training would not introduce a new or different kind of accident.] As previously noted, Palo Verde's licensed operator training program is accredited by INPO and is based on a systems approach to training.

Additionally, the proposed TS change does not affect plant design, hardware, system operation, or procedures. Thus, the proposed amendment request 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 TS change is an administrative change to clarify the current requirements applicable to licensed operator qualifications and licensed operator-training program. This change is consistent with the requirements of 10 CFR [Part] 55. The TS qualification requirements for all other unit staff remain unchanged.

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

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

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

Therefore, there is no change in the analysis results and the proposed amendment request does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Kenneth C. Manne, Senior Attorney, Arizona Public Service Company, P.O. Box 52034, Mail Station 7636, Phoenix, Arizona 85072–2034.

NRC Section Chief: Stephen Dembek.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: December 10, 2002.

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.6.E, ''Jet Pump Surveillance Requirements'' and its Bases. Specifically, Notes 1 and 2 would be added to the surveillance to provide clarity for performing the surveillance under the designated condition. The proposed change would also modify the applicability of the surveillance. Additionally, the condition for flow imbalance of the two recirculation loops would be changed from 15% to 10%. A reference in TS 4.11.C.1 to the bases for Specification 3.3.B.5 would also be changed to reference TS Table 3.2.C.1, Note 5.

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

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

Response: No.

The proposed Pilgrim TS 4.6.E imposes more restrictive surveillance requirements in accordance with the Standard Technical Specifications (STS) surveillance requirement 3.4.3.1 to ensure jet pump integrity during startup and run modes. The more restrictive conditions are: the recirculation loops have a flow imbalance of less than 10%, instead of the current 15%, when the pumps are operated at the same speed, and the occurrence of two of three conditions, instead of the simultaneous occurrence of all three conditions currently specified in TS 4.6.E for jet pump integrity.

The proposed more restrictive surveillance requirements ensure safe operation of the plant during startup and run modes. The requirements are not accident precursors. The proposed change that corrects a reference in Surveillance 4.11.C.1 is an administrative change with no impact on safety. These changes do not create accident conditions or increase the probability of previously evaluated accidents. The proposed changes provide additional assurance that the assumptions (*i.e.*, jet pump integrity) are met. Therefore, the probability or the consequences of an accident previously evaluated are not significantly increased.

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

Response: No.

The proposed changes do not involve a change to the plant design or a new mode of equipment operation. As a result, the proposed changes do not affect parameters or conditions that could contribute to the initiation of any new or different kind of accident. Therefore, these proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed surveillance requirements increase the margin of safety by providing additional assurance of jet pump integrity. The proposed change to correctly reference the existing Specification is administrative in nature. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

Entergy Nuclear Operations, Inc., Docket No. 50–293, Pilgrim Nuclear Power Station, Plymouth County, Massachusetts

Date of amendment request: March 19, 2003.

Description of amendment request: The proposed amendment would delete Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' requirements 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 an 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 changes are based on NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF–413, ''Elimination of Requirements for a Post Accident Sampling System (PASS).'' The U.S. Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on December 27, 2001 (66 FR 66949), on possible amendments concerning TSTF–413, 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 March 20, 2002 (67 FR 13027). The licensee affirmed the applicability of the following NSHC determination in its application dated March 19, 2003.

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

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

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

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

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

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

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

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

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

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

NRC Section Chief: James W. Clifford.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 8, 2003.

Description of amendment request: The proposed amendment would change Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to add a note allowing intermittent opening of penetration flow paths, under administrative control, that are isolated to comply with TS ACTIONS and to revise the operability requirement for the Reactor Core Isolation Cooling (RCIC) steam supply line low pressure isolation instrumentation to be consistent with the RCIC system operability requirements.

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

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

Response: No.

The proposed change to adopt TSTF [Technical Specification Task Force]-306 allows primary containment and drywell isolation valves to be unisolated under administrative controls when the associated isolation instrumentation is not operable. The isolation function is an accident mitigating function and is not an initiator of an accident previously evaluated. Administrative controls are required to be in effect when the valves are unisolated so that the penetration can be rapidly isolated when the need [for isolation] is indicated. Therefore the probability or consequences of previously evaluated accidents are not significantly increased.

The proposed change also allows the RCIC turbine steam line low pressure containment isolation instrumentation to be inoperable during low startup operating pressures. These instruments primarily provide automatic isolation when steam line pressure is too low for RCIC turbine operation. The low pressure automatic isolation feature will only be unavailable during the time that the RCIC system is not required to be operable. Therefore the change does not adversely affect the ability of the RCIC system to perform its safety function.

The RCIC steam line low pressure instruments also provide a diverse signal to indicate a possible system break. Even though the low pressure automatic isolation function will not be available for a short period during plant startup, the likelihood of a steam line break during the short period of time is low due to the low operating pressure. In addition, the safety function of providing containment integrity is maintained since there are other diverse leak detection instruments as well as other barriers or isolation capabilities that provide the isolation function.

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

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

Response: No.

The proposed change does not involve any physical alteration of plant equipment and does not change the method by which any safety-related system performs its function. The TS currently allow[s] containment and drywell isolation valves to be open under administrative controls after being closed to comply with TS ACTIONS for inoperable valves. Extending this allowance to the supporting instrumentation does not introduce any new method of isolation that has not already been evaluated.

Allowing the RCIC turbine steam line low pressure isolation instrumentation to be inoperable during low startup operating pressures does not create the possibility of any new failure modes other than those previously evaluated. No new or different type of equipment will be installed. There are no new failure mechanisms or accident initiators introduced. The low pressure isolation is designed to terminate RCIC turbine operation at low steam pressures for equipment protection. However, this function is not required since the RCIC system is not required to be operable and the same function is accomplished by maintaining the turbine trip/throttle valve closed. The low pressure isolation function will continue to be required when the RCIC system is required to be operable.

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

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

Response: No.

The change to allow containment and drywell isolation valves to be unisolated under administrative control does not reduce any margins to safety since the proposed allowance for the supporting isolation instrumentation is no less restrictive than the allowance for the equipment it supports. When the valves are unisolated, the design basis function of containment isolation is maintained by administrative controls.

The change to allow the RCIC turbine steam line low pressure isolation instrumentation to be inoperable during low startup operating pressures does not reduce any margins to safety. The current bounding analysis for a steam line break outside of containment remains bounding for a[n] RCIC steam break at lower pressures. In addition, the current high energy line break evaluations and subcompartment pressurization evaluations remain bounding for the low pressure condition. The design basis functions of containment isolation and containment integrity are maintained by the diverse leak detection instruments as well as other barriers or isolation capabilities that provide the isolation function.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 12, 2003.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to remove the MODE restrictions for performance of Surveillance Requirement (SR) 3.8.4.7 and SR 3.8.4.8 for the Division 3 direct current electrical power subsystem.

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

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

Response: No.

The power supplied by the battery is used only as a source of control and motive power for the HPCS [High Pressure Core Spray] system logic, HPCS diesel-generator set control and protection, and other Division 3 related controls. The loads supplied by this system are only loads associated with Division 3 of the Emergency Core Cooling Systems (ECCS).

The battery testing period is within the period of time that the system is scheduled to be out of service for other planned maintenance. The battery test does not increase unavailability of the supported system or represent any change in risk above the current practice of planned system maintenance outages as currently allowed by the TS. Any risk associated with the testing of the Division 3 batteries will be enveloped by the risk management of the system outage.

The out of service condition is controlled and evaluated for safety implications in accordance with 10 CFR 50.65. The HPCS system reliability and availability are monitored and evaluated in relationship to Maintenance Rule goals to ensure that total outage times do not degrade operational safety over time.

Therefore, the proposed change will have no effect on the probability or consequences of any previously evaluated accident.

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

Response: No.

This request involves the testing of the HPCS battery on-line while the system is already out of service. The testing will not add additional out of service time. Testing during this period has no influence on, nor does it contribute in any way to, the possibility of a new or different kind of accident or malfunction from those previously analyzed. The method of performing the test is not changed. No new accident modes are created by testing during the period when the system is already unavailable. Because the system is already out of service, no safety-related equipment or safety functions are altered as a result of this change.

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

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

Response: No.

The battery testing will be performed when the HPCS system is already out of service for maintenance. The out of service condition is controlled and evaluated for safety implications in accordance with 10 CFR 50.65. The batteries are not expected to be unavailable for more than 24 hours. This testing period is within the period of time that the system is scheduled to be out of service for other planned maintenance. Therefore, the battery test does not increase unavailability of the supported system or represent any change in risk above the current practice of planned system maintenance outages as currently allowed by the TS. Timing of this test has no effect on any fission product barrier.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 12, 2003.

Description of amendment request: The proposed amendment would change the Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation,'' to add a provision to the APPLICABILITY requirement specified in Table 3.3.6.1–1, to eliminate the requirement that the instrumentation for the Residual Heat Removal (RHR) System Isolation

Function on Reactor Vessel Water Level-Low, Level 3, be OPERABLE during certain conditions in MODE 5. Specifically, the proposed change would remove the requirement when the upper containment reactor cavity is at the High Water Level condition specified in TS 3.5.2, ''Emergency Core Cooling Systems (ECCS) Shutdown.''

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

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

Response: No.

The proposed change revises the applicability requirement for the Residual Heat Removal (RHR) System Isolation function of the Primary Containment and Drywell Isolation Instrumentation during MODE 5. The change removes the requirement that the instrumentation be operable during certain conditions during refueling outages. The function is intended to mitigate reactor vessel draindown events. Although draindown events during refueling operations are not specifically evaluated in the Updated Final Safety Analysis Report (UFSAR), these events were evaluated in support of licensing actions for the Alternate Decay Heat Removal System (ADHRS). The probability that a draindown event will be initiated is unrelated to operability requirement for this instrumentation or the associated isolation valves. The evaluation supporting this change determined that mitigating actions can be taken to terminate all postulated draindown events prior to fuel uncovery. As a result, the probability of draindown events causing fuel uncovery and the potential for radiological releases has not significantly increased. The operation or failure of the shutdown cooling suction isolation does not contribute to the occurrence of an accident. No active or passive failure mechanisms that could lead to an accident are affected by the proposed change.

The consequences of a vessel drainage event are not significantly increased by the proposed change. Entergy [Entergy Operations, Inc.] has evaluated various draindown and pumpdown events through the shutdown cooling flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate such events such that adequate core cooling is maintained and a radiological release does not occur.

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

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

Response: No.

Entergy has evaluated various draindown events through the shutdown cooling flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate any events such that adequate core cooling is maintained. With the containment refueling cavity flooded, sufficient inventory is available to allow operator action to terminate the inventory loss prior to reaching a low water level in the reactor. Installed equipment is not operated in a new or different manner, no new or different system interactions are created, and no new processes are introduced. No new failures have been created by the proposed changes.

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

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

Response: No.

The proposed changes do not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the containment and drywell isolation function is also unchanged. The change simply modifies the applicability of the Technical Specifications (TS) by removing the requirement that the RHR system isolation on low reactor vessel level be operable with the upper containment cavity flooded in MODE 5. During MODE 5, the RHR system isolation mitigates postulated draindown events through the RHR system. Entergy has evaluated various draindown events through this flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate such events such that adequate core cooling is maintained.

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: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50–416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: May 12, 2003.

Description of amendment request: The proposed amendment would change administrative Technical Specification (TS) 5.5.12 regarding containment integrated leakage rate testing (ILRT) and TS 3.6.5.1.1 regarding drywell bypass leak rate testing (DWBT). The change would allow for a one-time extension of the interval (15 years) for performance of the next ILRT and DWBT.

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

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

Response: No.

The proposed amendment to TS 5.5.12 adds a one-time extension to the current interval for Type A testing (*i.e.,* the ILRT) and the DWBT. The current interval of ten years, based on past performance, would be extended on a one-time basis to 15-years from the date of the last test. The proposed extension to the Type A test cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since research documented in NUREG–1493, ''Performance Based Containment Leak Rate Test Program,'' has found that, generically, fewer than 3% of the potential containment leak paths are not identified by Type B and C testing. A risk evaluation of the interval extension for GGNS [Grand Gulf Nuclear Station, Unit 1] is consistent with these results. In addition, the testing and containment inspections also provide a high degree of assurance that the containment will not degrade in a manner detectable only by a Type A test. Inspections required by the Maintenance Rule (10 CFR 50.65) and by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leak tightness.

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

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

Response: No.

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

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

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

Response: No.

The generic study of the increase in the Type A test interval, NUREG–1493, concluded there is an imperceptible increase in the plant risk associated with extending the test interval out to twenty years. The evaluations done in support of this change confirm that (conclusion). Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detects 97% of potential leakage paths. For the requested change in the GGNS ILRT/DWBT interval, it was determined that the risk contribution of leakage will increase 0.99%. This change is considered very small and does not represent a significant reduction in the margin of safety.

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

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

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, NW., 12th Floor, Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 18, 2003.

Description of amendment request: The proposed amendments would revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change will modify TS Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation,'' to add the requirement to perform a Channel Check in accordance with Surveillance Requirement (SR) 3.3.6.1.1 to thirteen listed instrument functions. The proposed change is the result of the replacement of existing plant equipment with equipment that has the capability of permitting the performance of a Channel Check with the plant in MODE 1, 2, and 3. The proposed change is consistent with the wording specified in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001.

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 probability or consequences of an accident previously evaluated.

The proposed change to Technical Specifications (TS) Table 3.3.6.1–1, ''Primary Containment Isolation Instrumentation'' will incorporate into the LaSalle County Station (LSCS) TS, wording specified in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001. The proposed change will modify TS Table 3.3.6.1–1 to add the requirement to perform a Channel Check in accordance with Surveillance Requirement (SR) 3.3.6.1.1 to thirteen listed instrument functions. The performance of TS surveillance testing is not a precursor to any accident previously evaluated. A Channel Check is a monitoring activity that does not represent an accident initiator. Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The function of instrumentation listed on TS Table 3.3.6.1–1, in combination with other accident mitigation features, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. The surveillance testing specified in TS Table 3.3.6.1–1 will provide assurance that the instrumentation will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The failure modes of the new instrumentation do not give rise to a new or different kind of accident. The proposed change does not introduce any new modes of system operation or failure mechanisms.

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

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

The leak detection system at LaSalle County Station uses ambient or differential temperature increases to detect small primary coolant boundary leaks in the Main Steam Line Tunnel and in various rooms of the Reactor Core Isolation Cooling (RCIC) System and the Reactor Water Cleanup (RWCU) System. The existing thermocouple monitors did not have the capability to allow a Channel Check to be performed without undue risk of initiating an inadvertent system isolation in MODE 1, 2 and 3. Thus, the LSCS TS took exception to the guidance contained in NUREG–1434 and did not specify on TS Table 3.3.6.1–1 that a SR 3.3.6.1.1 Channel Check be performed on the above listed thirteen instrument functions.

The new thermocouple monitors have continuously reading digital displays that permit the performance of a Channel Check with the Unit in MODE 1, 2 and 3 without risk of inadvertent system isolations. The new thermocouple digital displays have been installed on Unit 2 during the January/February 2003 refuel outage and are scheduled to be installed in Unit 1 during the upcoming January 2004 refuel outage. LSCS after the return to service of Unit 2 in March of 2003, verified that the thermocouple digital displays do permit a Channel Check to be successfully performed on the above listed thirteen instrument functions. Therefore, LSCS is requesting that TS Table 3.3.6.1–1 is modified to specify that a SR 3.3.6.1.1 Channel Check be performed in MODE 1, 2 and 3, consistent with the guidance contained in NUREG–1434, Rev. 2.

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

Based upon the above, Exelon Generation Company concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

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

Date of amendment request: April 18, 2003.

Description of amendment request: The proposed amendments would revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the proposed change will modify TS Surveillance Requirement (SR) 3.6.1.3.8 to identify that the specified testing requirement is applicable to reactor instrumentation lines. The proposed change is consistent with the SR wording specified in

NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 2, dated June 2001.

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 probability or consequences of an accident previously evaluated.

The proposed change to the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 will incorporate into the SR, wording specified in NUREG–1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' Revision 2, dated June 2001. The proposed change will specify that the testing required by SR 3.6.1.3.8 is applicable to reactor instrumentation line excess flow check valves (EFCVs). The performance of TS surveillance testing is not a precursor to any accident previously evaluated. Thus, the proposed change does not have any affect on the probability of an accident previously evaluated.

The function of reactor instrumentation line EFCVs, in combination with other accident mitigation features, is to limit fission product release. The surveillance testing specified in SR 3.6.1.3.8 will provide assurance that the reactor instrumentation line EFCVs will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

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

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

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

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

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

NUREG–1433, Rev. 2, provided licensees with the latest NRC recommended content and format for TS. The NUREG–1433 SR for testing EFCVs, SR 3.6.1.3.10, specifies that this testing is associated with reactor instrumentation line EFCVs. The Bases to SR 3.6.1.3.10 in NUREG–1433, Rev. 2, provides a reference to NEDO–32977–A, ''Excess Flow Check Valve Testing Relaxation,'' dated June 2000. NEDO–32977–A was approved for use by licensees in a NRC letter dated March 14, 2000. NEDO–32977–A states the following on the scope of TS testing associated with EFCVs:

EFCVs in instrument lines which connect to the reactor coolant pressure boundary (RCPB) are normally tested during refueling outages to meet Technical Specification requirements. Instrument lines that connect to the containment atmosphere, such as those which measure drywell pressure, or monitor the containment atmosphere or suppression pool water level, are considered extensions of primary containment. A failure of one of these instrument lines during normal operation would not result in the closure of the associated EFCV, since normal operating containment pressure is not sufficient to operate the valve. Such EFCVs will only close with a downstream line break concurrent with a Loss of Coolant Accident (LOCA). Since these conditions are beyond the plant design basis, EFCV closure is not needed and containment atmospheric instrument line EFCVs need not be tested.

The proposed change will incorporate the wording from NUREG–1433 into LaSalle County Station SR 3.6.1.3.8 to limit the scope of TS required testing to EFCVs that are directly connected to the RCPB.

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

Based upon the above, Exelon Generation Company concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: Mr. Edward J. Cullen, Deputy General Counsel, Exelon BSC—Legal, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: May 14, 2003.

Description of amendment request: The proposed amendment would modify the Technical Specifications by allowing entry into Mode 3 operation (shutdown with reactor coolant system temperature equal to or greater than 280 degrees Fahrenheit) during the current outage only with neither high pressure injection (HPI) pump capable of taking suction from the low pressure injection system trains when aligned for containment sump recirculation. The HPI system will otherwise be operable.

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

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

Response: No.

The proposed change allows the plant to operate in Mode 3 in support of RCS [reactor coolant system] leakage inspection activities conducted during the ongoing Thirteenth Refueling Outage, utilizing a limited exception to Limiting Condition for Operation (LCO) 3.5.2. This LCO applies in plant operational Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby). Under the proposed exception, for entry into Mode 3, both HPI trains would be required to be operable except for the capability of maintaining suction from the containment emergency sump during the recirculation phase.

The ability of the HPI pumps to draw suction from the containment emergency sump (via the LPI [low pressure injection] pumps) is a design feature credited by the Davis-Besse Nuclear Power Station Updated Safety Analysis Report (USAR) for mitigation of various types of loss-of-coolant accidents (LOCAs). Due to the potential susceptibility to damage from debris contained in the pumped fluid, the existing HPI pumps may not be capable of maintaining suction from the containment emergency sump without an increased probability for malfunction. However, the current plant conditions are unique in that decay heat generation rate in the reactor core is extremely low due to the fact that the plant has not operated in more than 14 months and 76 unirradiated fuel assemblies have been loaded into the core, replacing irradiated fuel assemblies.

A LOCA evaluation has been performed considering the current reactor core decay heat generation rate. The evaluation shows that in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase, while crediting additional operator actions not presently credited in the USAR. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for the additional operator actions, is very small. Also, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, radiological consequences would be very small compared to the accident analyses results of record, given the fission product decay over the extended plant shutdown. Therefore, the proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

There are no new or different accident initiators introduced by the proposed change to allow the plant to operate n Mode 3 under a limited exception, with the HPI pumps not capable of maintaining suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a LOCA. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed change allows the plant to operate in Mode 3 under a limited exception, with the HPI pumps not capable of maintaining suction from the containment emergency sump (via the LPI pumps) during the recirculation phase of a LOCA. Although the ability of the HPI pumps to draw suction from the containment emergency sump (via the LPI pumps) is a design feature credited by the Davis-Besse Nuclear Power Station USAR for mitigation of various types of LOCAs, an evaluation shows that given the extremely low decay heat generation rate in the reactor core under current plant conditions, and crediting additional operator actions, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for the additional operator actions, would be expected to be very small. Also, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, radiological consequences would be very small compared to the accident analyses results of record, given the fission product decay over the extended plant shutdown. Accordingly, given that accident severity or consequences will not be significantly increased under the proposed change, a significant reduction in a margin of safety is not involved.

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

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

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: May 19, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by removing the designation of safety grade as a description of the flow indication for the motor driven feedwater pump system. The licensee inadvertently requested that the flow indication be designated as safety grade in an amendment request that was approved as license Amendment No. 193.

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

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

Response: No.

The proposed change corrects a post modification and repair Surveillance Requirement for the Motor Driven Feedwater Pump System. This surveillance is not an initiator to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The Technical Specifications continue to require the MDFP System to be operable and capable of performing its design function. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

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

Response: No.

The proposed correction does not result in a significant reduction in the margin of safety. The corrected Surveillance Requirement continues to ensure that the Motor Driven Feedwater Pump System can perform its required function. Thus, appropriate equipment continues to be tested in a manner that provides confidence that the equipment can perform its assumed function. 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: Mary E. O'Reilly, Attorney, FirstEnergy Corporation, 76 South Main Street, Akron, OH 44308.

NRC Section Chief: Anthony J. Mendiola.

FirstEnergy Nuclear Operating Company, Docket No. 50–346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio

Date of amendment request: May 21, 2003.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) by relocating to the licensee's Technical Requirements Manual the TS surveillance requirement pertaining to flow balance testing of the emergency core cooling system (ECCS) high pressure injection and low pressure injection subsystems following system modifications that alter subsystem flow characteristics. Also, the proposed amendment would add an ECCS pump operability requirement to the TS consistent with NUREG–1430, Standard Technical Specifications-Babcock and Wilcox Plants, Revision 2.

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

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

Response: No.

The proposed surveillance requirement relocation and replacement does not alter the design, operation, or testing of any structure system or component. No previously analyzed accident scenario is changed. Initiating conditions and assumptions remain as previously analyzed. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed surveillance requirement relocation and replacement does not alter the design, operation, or testing of any structure system or component. No new or different accident initiators are created as a result of the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The proposed surveillance requirement relocation and replacement does not reduce or adversely affect the capabilities of the ECCS. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Anthony J. Mendiola.

Nuclear Management Company, LLC, Docket No. 50–305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin; Docket No. 50–255, Palisades Plant, Van Buren County, Michigan; and Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of amendment request: April 30, 2003.

Description of amendment request: The proposed amendments would revise the Kewaunee Nuclear Power Plant Technical Specification (TS) Section 6.3, ''Plant Staff Qualifications,'' Palisades Plant TS Section 5.3, ''Plant Staff Qualifications,'' and Point Beach Nuclear Plant TS 5.3, ''Facility Staff Qualifications,'' to specify an exception to the current TS minimum qualifications. This exception requires licensed operators to meet the education and experience eligibility requirements of the National Academy for Nuclear Training (NANT) (ACAD 00–003), ''Guidelines for Initial Training and Qualification of Licensed Operators,'' dated January 2000.

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

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

Response: No.

The proposed Technical Specification (TS) amendments are administrative changes to clarify the current requirements for licensed operator qualifications and licensed operator training program. With these amendments, the TS continue to meet the current requirements of 10 CFR 55.

Although licensed operator qualifications and training may have an indirect impact on accidents previously evaluated, the Nuclear Regulatory Commission (NRC) considered this impact during the rulemaking process, and by issuance of the revised 10 CFR 55 rule, concluded that this impact remains acceptable, as long as the licensed operator training programs are certified to be accredited and are based on a systems approach to training. NMC licensed operator training programs are accredited by the National Nuclear Accrediting Board (NNAB) and are based on a systems approach to training. The proposed TS amendments take credit for the NNAB accreditation of the licensed operator training programs. The TS requirements for all other facility staff qualifications remain unchanged.

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

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

Response: No.

The proposed TS amendments are administrative changes to clarify the current requirements for licensed operator qualifications and licensed operator training programs and to conform to the revised 10 CFR 55.

As discussed above, although licensed operator qualifications and training may have an indirect impact on the possibility of a new or different kind of accident from any accident previously evaluated, the NRC considered this impact during the rulemaking process, and by issuance of the revised rule, concluded that this impact remains acceptable, as long as licensed operator training programs are certified to be accredited and based on a systems approach to training. As previously noted, NMC licensed operator training programs are accredited by NNAB and are based on a systems approach to training. The proposed TS amendments take credit for the NNAB accreditation of the licensed operator training programs. The TS requirements for all other facility staff qualifications remain unchanged.

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

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

Response: No.

The proposed TS amendments are administrative changes to clarify the current requirements applicable to licensed operator qualifications and licensed operator training programs. With these changes the TS continue to be consistent with the requirements of 10 CFR 55. The TS qualification requirements for all other facility staff remain unchanged.

Licensed operator qualifications and training can have an indirect impact on a margin of safety. However, the NRC considered this impact during the rulemaking process, and by issuance of the revised 10 CFR 55, determined that this impact remains acceptable, when licensees maintain a licensed operator training program that is accredited and based on a systems approach to training. As noted previously, NMC licensed operator training programs are accredited by NNAB and are based on a systems approach to training.

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

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

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

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

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: May 22, 2003.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant (KNPP) operating license and Technical Specifications (TSs) to increase the licensed rated power by 6.0 percent from 1673 megawatts thermal (MWt) to 1772 MWt.

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:

Proposed Power Level Changes

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

The stretch uprate evaluations performed included performance of accident analyses at uprated power parameters using approved methodologies. Results of these analyses continue to meet the event acceptance criteria. An evaluation of components and systems, including interface and control systems, that could be affected by the change in power level, were performed for the stretch power uprate. Components and systems will continue to function as designed and performance requirements for these systems will continue to be met. Additionally, the proposed change in power level was not found to initiate any accident, and therefore, does not increase the probability of an accident.

Dose consequences were evaluated using the uprated power parameters. Acceptance criteria continue to be met. Therefore, the change also does not increase the consequences of an accident.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of this proposed change in power. Therefore, the possibility of a new or different kind of accident is not created.

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

All analyses supporting the proposed uprated power condition continue to meet the appropriate acceptance criteria. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the changes in rated power level.

Proposed Safety Limit Change

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

The proposed change is an industry accepted safety limit applicable to the KNPP transition to Westinghouse fuel. Therefore, the change does not increase the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed change in fuel centerline temperature. The change has no adverse effect on the fuel or the performance or integrity of the fuel. Therefore, the possibility of a new or different kind of accident is not created.

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

The proposed safety limit change is backed by technical evaluations performed by Westinghouse and experimental data. The limit is shown to be met as part of reload safety evaluations performed on a cycle specific basis. All applicable analyses supporting the proposed uprated power condition continue to meet the appropriate acceptance criteria. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the change in the safety limit.

Engineered Safety Feature (ESF) Setting Change

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

The stretch power uprate evaluations performed included performance of accident analyses. Results of the accident analyses have verified that the acceptance criteria continue to be met. Neither the change in the analytical limit nor the change in the TS setting limit changes how the system functions. Systems will continue to function as designed and system performance criteria will continue to be met. Dose consequences have also been evaluated at uprate conditions and doses remain within the appropriate acceptance criteria. Therefore, the change does not increase the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of the proposed change in the high-high steam flow TS setting limit. Therefore, the possibility of a new or different kind of accident is not created.

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

The results of the accident analyses demonstrate the acceptance criteria continue to be met. Systems will continue to function as designed and system performance criteria continue to be met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the change in the high-high steam flow TS setting limit.

Proposed Containment Cooling Systems Change

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

Removal of the LCO [limiting condition for operation] is conservative in that it eliminates relaxation of a design requirement for system redundancy. Deletion of the less conservative condition is more conservative by definition. Maintaining the system in a more conservative condition cannot create new challenges to components and systems that could adversely affect their ability to mitigate accident consequences or diminish the integrity of any fission product barrier. Therefore, the deletion of the LCO does not increase the probability or consequences of an accident previously evaluated.

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

Maintaining the system in a more conservative condition does not adversely affect any fission product barrier, nor does it alter the safety function of safety related systems, structures, and components depended upon for accident prevention or mitigation. Equipment important to safety will continue to function at its design capacity. No new equipment is being added, replaced, or taken away by the deletion of the LCO. Therefore, the possibility of a new or different kind of accident is not created.

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

Safety analysis acceptance criteria continue to be satisfied for containment heat removal with deletion of this LCO. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the containment cooling systems change.

Proposed Condensate Storage Tank (CST) Changes

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

The stretch power uprate project evaluations performed included a review of the SBO [station blackout] event. Results of the evaluation verified that with the increase in the CST [condensate storage tanks] inventory, the evaluation criteria continue to be met. Systems will continue to function as designed and system performance criteria will continue to be met. Additionally, dose consequences have been evaluated for the power uprate and results remain within the appropriate acceptance criteria. Therefore, the changes to CST inventory do not increase the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The changes have no adverse effect on any safety related system and do not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of the proposed changes in inventory. Therefore, the possibility of a new or different kind of accident is not created.

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

The results of the SBO event review have verified that the analysis criteria continue to be met. Systems will continue to function as designed and system performance criteria continue to be met. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the changes in CST inventory.

Proposed Auxiliary Feedwater System Changes

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

The LONF accident analyses have demonstrated that the TS required AFW [auxiliary feedwater] trains at the minimum assumed flow capability provide sufficient heat removal capacity to mitigate the LONF accident such that acceptance criteria are satisfied. Single failure criteria are still met, and no physical system changes have been made. Dose consequences have been evaluated for the power uprate and the results remain within the appropriate acceptance criteria. Therefore, the changes to the auxiliary feedwater system technical specifications do not increase the probability or consequences of an accident previously evaluated.

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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The change has no adverse effect on any safety related system and does not change the performance or integrity of any safety related system. Additionally, no new safety related equipment is being added or changed as a result of these proposed changes to technical specifications. Therefore, the possibility of a new or different kind of accident is not created.

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

The LONF analysis supporting the proposed changes to technical specifications meets the appropriate acceptance criteria. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the auxiliary feedwater system technical specification changes.

Proposed Editorial and Administrative Changes

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

The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. Therefore, the changes do not increase the probability or consequences of an accident previously evaluated.

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

The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed editorial and administrative changes. Therefore, the possibility of a new or different kind of accident is not created.

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

The editorial and administrative changes do not affect the analysis performed in support of the stretch power uprate. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, there are no significant hazards associated with the editorial changes.

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

Attorney for licensee: John H. O'Neill, Jr., Esq., Shaw Pittman, Potts & Trowbridge, 2300 N. Street, NW, Washington, DC 20037–1128.

NRC Section Chief: L. Raghavan.

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

Date of amendment request: April 10, 2003.

Description of amendment request: The amendments would modify the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2, Technical Specifications (TSs) Table 3.3–1 ''Condition and Setpoint'' description for permissive P–7 to reflect the new location of pressure transmitters.

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

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

The proposed change to replace the words ''impulse chamber'' with ''steam line input'' in the descriptive text associated with the P–7 function of the Reactor Trip System does not involve any physical or design change to the P–7 function. The proposed change renames the turbine inlet pressure to reflect the change in turbine design and the new location where the pressure is sensed. Because the P–7 function is not affected by the proposed amendment request, the changes to the Salem TSs are effectively editorial in nature. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The intent of the proposed change is to revise the description of the P–7 permissive as a result of changes to the design of the turbine. The P–7 permissive function is based on a relationship between first stage turbine inlet pressure and rated thermal power (RTP). Although the pressure sensed at the new location will be slightly higher, the instrument and controls logic, and all design basis functions that rely on the P–7 function, will remain the same. Therefore, the proposed change does not create the possibility of a new or different kind of accident than any previously evaluated.

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

As previously stated, the proposed change is editorial in nature and maintains the design basis functions associated with the P–7 permissive interlock. This is accomplished because the turbine pressure input to the P–7 function will continue to exhibit a consistent and accurate relationship to RTP following plant modifications. Therefore, because there will be no changes to the input assumptions associated with Salem's accident analysis, the proposed change does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: April 11, 2003.

Description of amendment request: The amendment would modify Surveillance Requirements and Bases regarding response time testing of the Engineered Safeguards System Actuation System (ESFAS) and the Reactor Trip System (RTS).

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

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

Response: No.

This change to the Technical Specifications does not result in a condition where the design, material, and construction standards that were applicable prior to the change are altered. The same RTS and ESFAS instrumentation is being used; the time response allocations/modeling assumptions in the Chapter 15 analyses are still the same; only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the SAR [safety analysis report]. Therefore, the proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

The proposed change to remove the footnote from Unit 1 Surveillance Requirement 4.3.2.1.3 is an administrative change and does not result in any increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

This change does not alter the performance of the pressure and differential pressure transmitters and switches used in the plant protection systems. All sensors will still have response time verified by test before placing the sensor in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain sensors (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the sensor response characteristic. Implementation of 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 remove the footnote from Unit 1 Surveillance Requirement 4.3.2.1.3 is an administrative change and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect sensor response time. Based on the above, it is concluded that the proposed license amendment does not result in a reduction in margin with respect to plant safety.

The proposed change to remove the footnote from Unit 1 Surveillance Requirement 4.3.2.1.3 is an administrative change and 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: Jeffrie J. Keenan, Esquire, Nuclear Business Unit—N21, P.O. Box 236, Hancocks Bridge, NJ 08038.

NRC Section Chief: James W. Clifford.

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

Date of amendment request: March 31, 2003.

Description of amendment request: The proposed change would replace ''Central Power and Light Company (CPL)'' with ''AEP Texas Central Company'' throughout the Operating License of each unit.

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

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

Response: No.

The proposed administrative license amendment only changes the name of one of the owners of STP in the Operating Licenses. This is not an initiator for accidents nor does this action affect the consequences of an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed administrative license amendment only changes the name of one of the owners of STP in the Operating Licenses. This is not an initiator for accidents. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

Margin of safety is associated with confidence in the ability of the fission product barriers (*i.e.*, fuel and fuel cladding, reactor coolant pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed administrative license amendment only changes the name of one of the owners of STP in the Operating Licenses. The proposed action does not affect margin of safety at all. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment involves no significant hazards consideration under the standards set forth in 10 CFR 50.92 and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

Attorney for licensee: A. H. Gutterman, Esq., Morgan, Lewis & Bockius, 1111 Pennsylvania Avenue, NW., Washington, DC 20004.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: April 30, 2003.

Description of amendment request: The amendment would modify several surveillance requirements (SRs) in Technical Specifications (TSs) 3.8.1 and 3.8.4 on alternating current and direct current sources, respectively, for plant operation. The revised SRs would have notes deleted or modified to allow the SRs to be performed, or partially performed, in reactor modes that are currently not allowed by the TSs. The current SRs are not allowed to be performed in Modes 1 and 2. Several of the current SRs also cannot be performed in Modes 3 and 4.

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 design of plant equipment is not being modified by the proposed changes. In addition, the DGs [diesel generators] and their associated emergency loads are accident mitigating features. As such, testing of the DGs themselves is not associated with any potential accident-initiating mechanism. Therefore, there will be no significant impact on any accident probabilities by the approval of the requested changes.

The changes include an increase in the online time that a DG under test will be paralleled to the grid (for SRs 3.8.1.10 and

3.8.1.14). As such, the ability of the tested DG to respond to a design basis accident [(DBA)] could be adversely impacted by the proposed changes. However, the impacts are not considered significant based, in part, on the ability of the remaining DG to mitigate a DBA or provide safe shutdown. With regard to SR 3.8.1.10 and SR 3.8.1.14, experience shows that testing per these SRs typically does not perturb the electrical distribution system. In addition, operating experience and qualitative evaluation of the probability of the DG or bus loads being adversely affected concurrent with or due to a significant grid disturbance, while the DG is being tested, support the conclusion that the proposed changes do not involve any significant increase in the likelihood of a safety-related bus blackout or damage to plant loads.

The SR changes that are consistent with TSTF [Technical Specification Task Force]-283 have been approved by the NRC for submittal by licensees. The on-line tests allowed by the TSTF are only to be performed for the purpose of establishing OPERABILITY [of the DG being tested]. Performance of these SRs during restricted MODES will require an assessment to assure plant safety is maintained or enhanced.

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

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

The capability to synchronize a DG to the offsite source (via the associated plant bus) and test the DG in such a configuration is a design feature of the DGs, including the test mode override in response to a safety injection signal. Paralleling the DG for longer periods of time during plant operation may slightly increase the probability of incurring an adverse effect from the offsite source, but this increase in probability is judged to be still quite small and such a possibility is not a new or previously unrecognized consideration.

The proposed changes would not require any new or different accidents to be postulated since no changes are being made to the plant that would introduce any new accident causal mechanisms. This license amendment request does not impact any plant systems that are potential accident initiators; nor does it have any significantly adverse impact on any accident mitigating systems.

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 changes do not involve a significant reduction in the margin of safety. The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design [safety] functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The proposed changes do not directly affect these barriers, nor do they involve any significantly adverse impact on the DGs which serve to support these barriers in the event of an accident concurrent with a loss of offsite power. The proposed changes to the testing requirements for the plant DGs do not affect the OPERABILITY requirements for the DGs, as verification of such OPERABILITY will continue to be performed as required (except during different allowed MODES [of operation]). These changes have an insignificant impact on DG availability, as continued verification of OPERABILITY supports the capability of the DGs to perform their required [safety] function of providing emergency power to plant equipment that supports or constitutes the fission product barriers. Only one DG is to be tested at a time, so that the remaining DG will be available to safely shut down the plant if required. Consequently, performance of the fission product barriers will not be impacted by implementation of the proposed amendment.

In addition, the proposed changes involve no changes to [safety] setpoints or limits established or assumed by the accident analyses. On this and the above basis, no safety margins will be impacted.

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

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

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

NRC Section Chief: Stephen Dembek.

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

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

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

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

Date of amendment request: May 1, 2003, as supplemented by letter dated May 2, 2003.

Brief description of amendment request: The proposed amendments would modify Technical Specification Surveillance Requirements to provide an alternative means of testing the Unit 1 main steam electromatic relief valves, including those that provide the automatic depressurization and the low set relief functions, and provide an alternative means for testing the Units 1 and 2 dual function Target Rock safety/relief valves.

*Date of publication of individual notice in **Federal Register:*** May 13, 2003 (68 FR 25645).

Expiration date of individual notice: May 27, 2003.

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, Public File Area 01F21, 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/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

Detroit Edison Company, Docket No. 50–16, Enrico Fermi Atomic Power Plant, Unit 1, (Fermi 1) Monroe County, Michigan

Date of amendment request: January 28, 2003 (Reference NRC–03–0011).

Brief description of amendment: This amendment revises the Fermi 1, Technical Specifications by removing the requirements for Water Intrusion alarms, associated surveillances, and liquid waste tank level check surveillance. The sections containing Reactor Building and Fuel and Repair Building drains descriptions are removed in their entirety, clarification is added for evolutions when tritium sampling is not required. This amendment also removes previously deleted items and re-numbers/letters remaining sections, and makes several editorial corrections.

Date of issuance: May 16, 2003.

Effective date: On the date of issuance of this amendment and must be fully implemented no later than 60-calendar days from the date of issuance.

Amendment No.: 20.

Facility Operating License No. DPR–9: Amendment revised the Technical Specifications by: (1) Deleting Sections A.1, 2, 4, 8, C.1, D, E.1, H.3.b, I.5, I.7b, I.9.d, which were previously deleted and the word ''Deleted'' used as a place marker to alleviate the need to renumber or re-letter the remaining sections. Also, the remaining sections were renumbered or re-lettered as appropriate. (2) Deleting Sections C.2 and E.2 which cover the Reactor Building and Fuel and Repair Building Drains. These requirements are no longer necessary in this phase of Fermi 1 decommissioning. (3) In Section F, the following words were added, ''Monitoring or sampling for tritium will not be required if the sample results have determined that tritium is not present during a given evolution.'' This wording was added to clarify during which evolutions resulting in radioactive gaseous effluents the effluents would be monitored or sampled and analyzed for tritium. (4) Sections H.1 and H.2, which covered water intrusion monitoring system alarms, including surveillances, allowed out-of-service time, compensatory measures and alarm readouts for alarms associated with water intrusion, were deleted. (5) In Section H.3 the surveillance requirement for radiation for the sump pump serving the reactor building annulus will not be required once the pump is made inactive and the surveillance requirement for radiation of the steam cleaning room access plug is deleted. In Section H.4 the requirement for a monthly level check of the liquid waste tanks was deleted. (6) Table H–1, which only lists water intrusion alarms, was deleted. (7) Editorial changes included in this amendment are in Section I.2, the word ''employes'' was changed to ''employees'; in Section I.2.b the word ''He'' was changed to ''The Health Physicist'; in Section I.7 the word ''his'' was removed from the following sentence, ''The Custodian may temporarily change a procedure by Written Order following his determination that the change does not constitute a significant increase in the hazards associated with the operation.'' In Section I.9.h the word ''usual'' will be changed to ''unusual.''

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18271). The NRC's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2003.

No significant hazards consideration comments: None received.

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

Date of application for amendment: May 23, 2002.

Brief description of amendment: The amendment deletes License Condition 2.C.(19) of the Operating License which pertains to historical actions that have been met. The amendment also deletes Section 2.F of the Operating License which requires reporting violations of the requirements in Section 2.C of the Operating License. The reporting requirements in Section 2.F are either adequately addressed by the requirements of 10 CFR 50.72 and 10 CFR 50.73, or are not needed because more restrictive requirements are contained in the specific License Condition.

In its May 23, 2003, application, the licensee also proposed to delete License Conditions 2.C.(20) and 2.C.(21) which pertain to historical actions that have been met. The Nuclear Regulatory Commission staff's evaluation of the proposed deletion of License Conditions 2.C.(20) and 2.C.(21) will be addressed under separate cover.

Date of issuance: May 16, 2003.

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

Amendment No.: 155.

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

Date of initial notice in ***Federal Register:*** June 25, 2002 (67 FR 42817).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2003.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50–245, Millstone Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: May 13, 2002.

Brief description of amendment: The amendment revised selected radiological-related technical specifications of the Millstone Unit 1 Permanently Defueled Technical Specifications. These changes are a result of the revision to part 20 of title 10 of the Code of Federal Regulations.

Date of issuance: May 15, 2003.

Effective date: May 15, 2003, and shall be implemented within 120 days from the date of issuance.

Amendment No.: 112.

Facility Operating License No. DPR–21: The amendment revised the Permanently Defueled Technical Specifications.

Date of initial notice in ***Federal Register:*** July 23, 2002 (67 FR 48215). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: May 13, 2002.

Brief description of amendments: The amendments revise the Millstone Power Station, Unit No. 2 (MP2) and Unit No. 3 (MP3) Technical Specifications (TSs) changing selected MP2 and MP3 radiological-related TSs. These changes are due to the revision to part 20 of title 10 of the Code of Federal Regulations.

Date of issuance: May 15, 2003.

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

Amendment Nos.: 276 and 215.

Facility Operating License Nos. DPR–65 and NPF–49: These amendments revised the TSs.

Date of initial notice in ***Federal Register:*** July 9, 2002 (67 FR 45562). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 15, 2003.

No significant hazards consideration comments received: No.

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

Date of application of amendments: February 17, 2003.

Brief description of amendments: The amendments revised the Technical Specifications Surveillance Requirement 3.10.1.9 to increase the loading requirements for the Standby Shutdown Facility Diesel Generator from ≥ 3000 kW to ≥ 3280 kW.

Date of Issuance: May 19, 2003.

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

Amendment Nos.: 331, 331, and 332.

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

Date of initial notice in ***Federal Register:*** April 1, 2003 (68 FR 15759). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 27, 2003.

Brief description of amendment: The amendment deletes Technical Specification 5.5.3, ''Post Accident Sampling,'' and thereby eliminates the requirements to have and maintain the post accident sampling system for the James A. FitzPatrick Nuclear Power Plant.

Date of issuance: May 16, 2003.

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

Amendment No.: 278.

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

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18276). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 12, 2002, as supplemented on April 3, 2003 and May 2, 2003.

Brief description of amendment: The amendment revises the Facility Operating License and the Technical Specifications (TSs) to increase the licensed core thermal power level to 3114.4 megawatts (MWt), which is a 1.4% increase above the currently authorized power level of 3071.4 MWt. The power uprate is based on the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, ''ECCS Evaluation Models,'' to Part 50 of Title 10 of the Code of Federal Regulations. Specifically, the reduced uncertainty is obtained by using a more accurate measurement of feedwater flow. In addition, changes were made to TS Sections 1.1, 2.1, 2.3, 3.1, 3.4, 6.9, and the applicable TS Bases to account for the change in power level.

Date of issuance: May 22, 2003.

Effective date: May 22, 2003.

Amendment No.: 237.

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

Date of initial notice in ***Federal Register:*** January 7, 2003 (68 FR 00801). The April 3 and May 3, 2003, letters provided clarifying information that did not enlarge the scope of the original **Federal Register** notice or change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 22, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 22, 2002, as supplemented by letter dated March 13, 2003.

Brief description of amendment: The amendment allows for a one-time change to revise the steam generator in-service inspection frequency requirements in Technical Specification 4.4.5.3.a to allow a 40-month inspection interval after one inspection, rather than after two consecutive inspections, based on the results falling into the C–1 classification.

Date of issuance: May 28, 2003.

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

Amendment No.: 247.

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

Date of initial notice in ***Federal Register:*** December 24, 2002 (67 FR 78520). The March 13, 2003, supplemental letter provided clarifying information that did not change the scope of the original **Federal Register** notice or the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 28, 2003.

No significant hazards consideration comments received: No.

Exelon Generation Company, LLC, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois; Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: December 12, 2002.

Brief description of amendments: The amendments would add a new Surveillance Requirement to Technical Specification Section 3.7.5, ''Auxillary Feedwater (AF) System,'' which requires operation of the diesel-driven AF pump on a monthly frequency (*i.e.*, once every 31 days) for greater than or equal to 15 minutes.

Date of issuance: May 22, 2003.

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

Amendment Nos.: 132/127.

Facility Operating License Nos. NPF–37, NPF–66, NPF–72 and NPF–77: The amendments revised the Technical Specifications 3.7.5.

Date of initial notice in ***Federal Register:*** February 18, 2003 (68 FR 7817). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 28, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 27, 2002.

Brief description of amendments: These amendments deleted TS 5.5.3, ''Post Accident Sampling,'' and thereby eliminated the requirements to have and maintain the post accident sampling system for Peach Bottom Atomic Power Station, Units 2 and 3.

Date of issuance: May 22, 2003.

Effective date: As of the date of issuance, to be implemented within 180 days.

Amendments Nos.: 248 and 251.

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

Date of initial notice in ***Federal Register:*** January 21, 2003 (68 FR 2802). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 22, 2003.

No significant hazards consideration comments received: No.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Units 1 and 2, Berrien County, Michigan

Date of application for amendments: August 30, 2002 as supplemented by letters dated February 27, April 7, April 29, and May 2, 2003.

Brief description of amendments: The amendments revise the reactor trip system and engineered safety features actuation system surveillance requirements, increasing selected surveillance intervals for analog channels, logic cabinets, and reactor trip breakers. Additionally, the amendments revise the reactor trip system and engineered safety features actuation system surveillance requirements, increasing the completion time and bypass time for the reactor trip breakers.

Date of issuance: May 23, 2003.

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

Amendment Nos.: 277 and 260.

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

Date of initial notice in ***Federal Register:*** October 15, 2002 (67 FR 63695). The supplemental letters 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 amendments is contained in a Safety Evaluation dated May 23, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 11, 2003.

Brief description of amendment: The amendment changes the operating license by adding a paragraph authorizing the licensee to revise the updated final safety analysis report by deleting the notation that the Nuclear Regulatory Commission does not endorse the reactor building crane as single-failure-proof.

Date of issuance: May 16, 2003.

Effective date: As of the date of issuance and shall be implemented no later than the update of the final safety analysis report to be submitted in accordance with 10 CFR 50.71(e).

Amendment No.: 251.

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

Date of initial notice in ***Federal Register:*** April 15, 2003 (68 FR 18278). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 16, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendments: July 10, 2002, as supplemented May 9, 2003.

Brief description of amendments: The amendments change the Sequoyah Nuclear Plant (SQN) Technical Specifications (TSs) by modifying the requirements applicable when actions or other requirements direct suspension of activities that involve a positive reactivity change for the SQN TSs. The proposed change will remove the requirement to not make positive reactivity changes during certain conditions. The changes will permit limited positive reactivity changes that are necessitated by plant operations. These changes will limit the amount of reactivity changes to those that will continue to assure appropriate reactivity limits are met, either shutdown margin or refueling boron concentration, as appropriate.

Date of issuance: May 22, 2003.

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

Amendment Nos.: 285 and 274.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 6, 2002 (67 FR 50961). The supplemental letters provided clarifying information that was within the scope of the initial notice and 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 May 22, 2003.

No significant hazards consideration comments received: No.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendment: November 15, 2002, as supplemented February 28, 2003, March 14, 2003, and April 25, 2003.

Brief description of amendment: The Amendments revise the Technical Specification (TS) 3.7.1.3, ''Condensate Storage Water,'' Limiting Condition for Operation by increasing the required minimum amount of stored water from 190,000 gallons to 240,000 gallons. This change is being made to support the replacement steam generator requirements.

Date of issuance: May 27, 2003.

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

Amendment Nos.: 286 and 275.

Facility Operating License No. DPR–77 and DPR–79: Amendments revise the TSs.

Date of initial notice in ***Federal Register:*** February 4, 2003 (68 FR 5682). The supplemental letters provided clarifying information that was within the scope of the initial notice and 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 May 27, 2003.

No significant hazards consideration comments received: No.

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

Date of amendments request: August 19, 2002.

Brief description of amendments: The amendment revised Technical Specification Section 3/4.3.2, ''Engineered Safety Features Actuation System Instrumentation,'' to extend the interval between slave relay tests.

Date of issuance: May 19, 2003.

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

Amendment Nos.: Unit 1–152 ; Unit 2–140.

Facility Operating License Nos. NPF–76 and NPF–80: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** October 1, 2002 (67 FR 61685). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 19, 2003.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of June, 2003.

For the Nuclear Regulatory Commission.

John A. Zwolinski,

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

[FR Doc. 03–14277 Filed 6–9–03; 8:45 am]

BILLING CODE 7590–01–P

POSTAL RATE COMMISSION

[Docket No. MC2003–2; Order No. 1373]

Experimental Parcel Return Services

AGENCY: Postal Rate Commission.

ACTION: Notice and order.

SUMMARY: This document provides notice that the Postal Service has filed a request with the Commission seeking an expedited decision approving a two-year experiment testing bulk parcel return services. It briefly describes the proposal, which focuses primarily on the customer-to-merchant segment of retail transactions. The notice also addresses related terms and conditions, proposed rates, and eligibility for participation in the experiment. It identifies conference dates and deadlines for certain procedural steps in the initial stages of this case.

DATES: 1. June 18, 2003: notices of intervention, requests for a hearing, and comments on experimental status.

2. June 24, 2003: (optional) comments on discovery-related deadlines.

3. June 25, 2003: prehearing conference (2 p.m.).

4. June 27, 2003: responses to conditional motion for waiver of certain filing requirements.

ADDRESSES: Submit documents electronically via the Commission's Filing Online system, which can be accessed at *http://www.prc.gov.* Settlement and prehearing conferences will be held in the Commission's hearing room, 1333 H Street NW., Suite 300, Washington, DC 20268–0001.

FOR FURTHER INFORMATION CONTACT: Stephen L. Sharfman, General Counsel, 202–789–6818.

SUPPLEMENTARY INFORMATION: On May 28, 2003, the Postal Service filed a request seeking a recommended decision approving an experimental change in the Domestic Mail Classification Schedule (DMCS) to establish rate categories, including rates and fees, for certain parcels and bound printed matter that are returns from customers to merchants.[1] The request, which includes six attachments, was filed pursuant to chapter 36 of the Postal Reorganization Act, 39 U.S.C. 3601 *et seq.*[2]

In contemporaneous filings, the Postal Service requests expedited consideration of its proposal, including establishment of settlement procedures,[3] and a conditional motion for waiver of the filing requirements.[4] The Postal Service's request for expedition is in addition to that generally available under the Commission's experimental rules [39 CFR 3001.67–3001.67d]. The request, accompanying testimony of witnesses Gullo (USPS–T–1), Eggleston (USPS–T–2), Kiefer (USPS–T–3), and Wittnebel (USPS–T–4), and other related material are available for inspection in the Commission's docket room during regular business hours. They also can be accessed electronically, via the Internet, on the Commission's Web site (*http://www.prc.gov*).

I. Proposed Parcel Return Services

The Postal Service proposes an experimental bulk parcel return service applicable to merchandise returned as either Parcel Post or Bound Printed Matter (BPM) mail. Collectively, the experimental changes are referred to as Parcel Return Services, comprised of Parcel Select Return Service (PSRS) and Bound Printed Matter Return Service (BPMRS). Witness Kiefer sponsors the proposed rates and classifications. *See* USPS–T–3. The proposed rates are based on workshare savings for returned parcels retrieved in bulk by shippers (or their agents) at designated delivery units or bulk mail centers.

PSRS adds two rate categories to the Parcel Post subclass, Parcel Select Return Delivery Unit (RDU) and Parcel Select Return Bulk Mail Center (RBMC). The proposed RDU rate for mail retrieved in bulk at delivery units is $2.00 per parcel. The proposed RBMC rates for parcels retrieved in bulk at the first BMC they reach range between $0.86 and $1.51 below the non-workshared rates for regular-sized parcels.[5] *Id.* at 2.

BPMRS adds one rate category to the BPM subclass, Bound Printed Matter Bulk Mail Center (RBMC). Similar to Parcel Select Return Service, the RBMC rate is applicable to BPM parcels retrieved in bulk at the first BMC they reach. The proposed rates are $0.24 below the non-workshared BPM rates. *Id.*; see also Request at 2.[6]

Witness Kiefer's proposed rates are based on cost data supplied by witness Eggleston. *See* USPS–T–2. The Postal Service indicates that the cost avoidance measures underlying its proposed rates are estimated using the same cost base supporting the Commission rate recommendations in Docket No. R2001–1. In addition, the Postal Service states that the proposed experiment will not materially affect its overall revenues. Request at 2–3.

In support of its proposal, the Postal Service also submits the testimony of witness Gullo (USPS–T–1), who describes the proposed Parcel Return Services products, and witness Wittnebel, who discusses, from a mailer's perspective, the processing of returns and the benefits associated with the experiment (USPS–T–4).

Experimental designation. By designating its proposal as experimental, the Postal Service seeks consideration of its Request under rules 67–67d. The Postal Service suggests that these rules are appropriate as they contemplate review of proposed experimental classifications in the absence of historical data that normally underlie requests for permanent classification changes. While acknowledging that it lacks data about the potential response to the

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[1] Request of the United States Postal Service for a Recommended Decision on Experimental Parcel Return Services, Docket No. MC2003–2, May 28, 2003 (Request).

[2] Attachment A contains the proposed classification schedule provisions; attachment B sets forth the proposed rate and fee schedules; attachment C contains the certified financial statements for the years ending September 30, 2001 and September 30, 2002; attachment D is the certification required by Commission rule 54(p); attachment E is an index of testimonies; and attachment F is the statement addressing compliance with various filing requirements.

[3] United States Postal Service Request for Expedition and Establishment of Settlement Procedures, May 28, 2003 (Request for Expedition).

[4] Statement of the United States Postal Service Concerning Compliance with Filing Requirements and Conditional Motion for Waiver, May 28, 2003 (Conditional Motion).

[5] Nonmachinable RBMC Parcel Post mail is subject to nonmachinable surcharges. See proposed DMCS 521.7.

[6] BPM mailers are eligible for RDU service and rates if they so choose.