Week of March 25, 2002—Tentative

There are no meetings scheduled for the Week of March 25, 2002.

*The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

The NRC Commission Meeting Schedule can be found on the Internet at: *www.nrc.gov*

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this commission meeting schedule electronically, please send a electronic message to *dkw@nrc.gov.*

Dated: February 14, 2002.

Sandra M. Joosten,

Executive Assistant, Office of the Secretary.

[FR Doc. 02–4027 Filed 2–14–02; 10:23 am]

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from January 25, 2002 through February 7, 2002. The last biweekly notice was published on February 5, 2002 (67 FR 5323).

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

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

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

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

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

By March 21, 2002, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the NRC's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/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, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, *http://www.nrc.gov/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, 304–415–4737 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: November 30, 2001.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001, (66 FR 32400), on possible amendments concerning missed surveillances, 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 September 28, 2001, (66 FR 49714). The licensees affirmed the applicability of the following NSHC determination in its application dated November 30, 2001.

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

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 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. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Anthony J. Mendiola.

Arizona Public Service Company, et al., Docket No. STN 50–529, Palo Verde Nuclear Generating Station, Unit 2, Maricopa County, Arizona

Date of amendment request: December 21, 2001.

Description of amendment request: The amendment would revise the operating license and the Technical Specifications (TSs) to support replacement of the steam generators and the subsequent increased power to a level of 3990 MWt, a 2.94 percent increase.

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.

a. Evaluation of the Probability of Previously Evaluated Accidents

Plant Structures, Systems and Components (SSCs) have been verified to be capable of performing their intended design functions at uprated power conditions. Where necessary, a small number of minor modifications will be made prior to implementation of uprated power operations so that surveillance test acceptance criteria continues to be met. The analysis has concluded that operation at uprated power conditions will not adversely affect the capability or reliability of plant equipment. Current technical specification surveillance requirements ensure frequent and adequate monitoring of system and component operability. All systems will continue to be operated within current operating requirements at uprated conditions. Therefore, no new structure, system or component interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR).

b. Evaluation of the Consequences of Previously Evaluated Accidents

The radiological consequences were reviewed for all design basis accidents (DBAs) (*i.e.,* both LOCA [loss-of-coolant accident] and non-LOCA accidents) previously analyzed in the UFSAR. The analyses showed that the resultant radiological consequences for both LOCA and non-LOCA accidents remained within regulatory and Standard Review Plan (SRP) limits at uprated power conditions.

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

The configuration, operation and accident response of the PVNGS [Palo Verde Nuclear Generating Station] Unit 2 SSCs are unchanged by operation at uprated power conditions or by the associated proposed TS changes. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident or different scenario.

The effect of operation at uprated power conditions on plant equipment has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as a result of operating at uprated conditions. In addition, operation at uprated power conditions does not create any new failure modes that could lead to a different kind of accident. Minor plant modifications, to support implementation of uprated power conditions, will be made as required to existing SSCs. The basic design function of all SSCs remains unchanged and no new equipment or systems have been installed that could potentially introduce new failure modes or accident sequences.

Based on these analyses, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not have an adverse effect on any safety-related system or design basis function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

A comprehensive analysis was performed to evaluate the effects of power uprate on PVNGS Unit 2. This analysis identified and defined the major input parameters to the NSSS [nuclear steam supply system], reviewed NSSS design transients, and reviewed the capabilities of the NSSS and BOP [balance-of-plant] fluid systems, NSSS/BOP interfaces, NSSS and BOP control systems, and NSSS and BOP SSCs. NSSS accident analyses were re-performed or reviewed to confirm that acceptable results were maintained and that the radiological consequences remained within regulatory and SRP limits. The nuclear and thermal hydraulic performance of nuclear fuel was also reviewed to confirm acceptable results. The analyses confirmed that all NSSS and BOP SSCs are capable, some with minor modifications, to safely support operations at uprated power conditions.

The margin of safety of the reactor coolant pressure boundary is maintained under uprated power conditions. The design pressure of the reactor pressure vessel and reactor coolant system will not be challenged as the pressure mitigating systems were confirmed to be sufficiently sized to adequately control pressure under uprated power conditions.

Reanalysis of containment structural integrity under DBA conditions indicates that the calculated peak containment pressure (Pa) increases from 52.0 psig to 58.0 psig, but remains less than the containment internal design pressure of 60 psig. The proposed value for Pa has been rounded up from the actual calculated value of 57.85 psig.

Radiological consequences of the following accidents were reviewed: Main Steam Line Break, Locked Reactor Coolant Pump (RCP) Rotor, CEA Ejection, Small Steam Line Break Outside Containment, Steam Generator Tube Rupture, LBLOCA [large break loss of coolant accident], SBLOCA [small break loss of coolant accident], Waste Gas Decay Tank Rupture, Liquid Waste Tank Failure, and Fuel Handling Accident. The resultant radiological consequences for each of these accidents remained within regulatory and SRP limits at uprated power conditions.

The analyses supporting operation at power uprate conditions have demonstrated that all systems and components are capable of safely operating at uprated power conditions. All DBA acceptance criteria will continue to be met. Therefore, it is concluded that 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 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 an amendment involves no significant hazards consideration.

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Section Chief: Stephen Dembek.

Calvert Cliffs Nuclear Power Plant, Inc., Docket No. 50–317, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of amendment request: January 31, 2002.

Description of amendment request: The proposed amendment would allow a one-time five-year extension, for a total of 15 years, for the performance of the next Unit 1 integrated leak rate test (ILRT). The proposed amendment would also exempt Unit 1 from the requirement to perform a post-modification containment ILRT associated with the steam generator replacement.

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

[Extension of Type A integrated leakage rate testing:]

This proposed one-time extension of the Type A test interval does not 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 of an accident since research documented in NUREG–1493 has found that, generically, fewer than three percent of the potential containment leak paths are not identified by Type B and C testing. Calvert Cliffs, through testing and containment inspections, also provides 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, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

[Exemption from post-modification testing:]

The steam generator replacement activities do not affect the containment structure or the actual containment liner. Access for the replacement steam generators as well as removal of the old steam generators will be through the equipment hatch. However, the outer shell of the steam generators, the inside containment portions of the main steam line, the feedwater lines, the auxiliary feedwater lines, and the steam generator blowdown lines are all part of the primary reactor containment boundary that will be impacted by the replacement activities.

Calvert Cliffs Nuclear Power Plant Technical Specification 5.5.16 states, ''A program shall be established to implement the leakage testing of the Containment as required by 10 CFR 50.54(o) and 10 CFR part 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, 'Performance-Based Containment Leak-Test Program' dated September 1995, including errata.'' Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' endorses NEI [Nuclear Energy Institute] 94–01, Revision 0 for methods acceptable to comply with the requirements of Option B. Prior to returning the Containment to operation, NEI 94–01 requires leakage rate testing (Type A testing or local leakage rate testing), following repairs and modification that affect the containment leakage integrity.

The affected area of the primary containment boundary is also part of the pressure boundary of an American Society of Mechanical Engineers (ASME) Class 2 component/piping system and, as such, the planned replacement of the steam generators are subject to the repair and replacement requirements of ASME Section XI. The ASME Section XI surface examination, volumetric examination, and system pressure test requirements are more stringent than the Appendix J, Option B testing requirements. The acceptance criteria for ASME Section XI system pressure testing of welded joints is ''zero leakage.'' In addition, the test pressure for the system pressure test will be approximately 17 times that of Appendix J, Option B test.

The objective of the Type A test is to assure the leak-tight integrity of the area affected by the modification. Although the leak test is in a direction reverse to that of the design basis accident environment, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Appendix J, Option B with the exception of secondary side access manways. Section 9.2.1, NEI 94–01, Revision 0 allows reverse testing if justified. Section XI pressure test applies a sealing pressure to the secondary manway due to the inward door swing configuration. Hence, a Type B local leak rate test will be performed for the secondary manways. For all other affected components, reverse testing is justified since the acceptance criteria for ASME Section XI system pressure testing of welded joints is ''zero leakage,'' and the test pressure of the system pressure test will be approximately 17 times that of a Type A test. Hence, the probability or consequences of design bases accidents previously evaluated are unchanged.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 1 steam generators will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

[Extension of Type A integrated leakage rate test interval:]

This proposed one-time 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 accident previously evaluated. The test itself is not changing and will be 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, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

[Exemption from post-modification testing:]

The proposed revision does not involve a physical change to the plant and there are no changes to the operation of the plant that could introduce a new failure mode. As described above in Item 1, the objective of the Appendix J, Option B test is to assure that leak-tight integrity of the area affected by the modification. The ASME Section XI inspection and testing requirements are more stringent than the Appendix J, Option B testing requirements.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 1 steam generators will not create the possibility of a new or different [kind] of accident from any previously evaluated.

3. Would not involve a significant reduction in the margin of safety.

[Extension of Type A integrated leakage rate test interval:]

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 20 years. Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detect 97 percent of potential leakage paths. For the requested change in the Calvert Cliffs Integrated Leakage Rate Test interval, it was determined that the risk contribution of leakage will increase 0.07 percent (based on change in offsite dose). This change is considered very small and does not represent a significant reduction in the margin of safety.

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

[Exemption from post-modification testing:]

As described above in Item 1, the ASME Section XI surface examination, volumetric examination, and system pressure test requirements are more stringent than the Appendix J, Option B testing requirements. The acceptance criteria for ASME Section XI system pressure testing of welded joints is "zero leakage." In addition, the test pressure for the system pressure test will be approximately 17 times that of Appendix J, Option B test.

Therefore, the proposed revision to Technical Specification 5.5.16 to eliminate the requirement to perform post-modification containment integrated leakage rate testing following replacement of Unit 1 steam generators does not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: Joel Munday (Acting).

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

Date of amendment request: November 6, 2001, as supplemented December 27, 2001.

Description of amendment request: The proposed amendment would: 1) Increase the allowable nominal average fuel assembly enrichment from 4.5 w/o U–235 to 4.85 w/o U–235 for all regions of the spent fuel pool, the new fuel storage racks (dry), and the reactor core; 2) Allow fuel to be located under the cell blockers in 40 empty Region B storage cells; and, 3) Credit spent fuel pool soluble boron for reactivity control during normal conditions to maintain spent fuel pool $K_{eff} \leq 0.95$.

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

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

Previously evaluated Final Safety Analysis Report (FSAR) Chapter 14 accidents are a fuel handling accident either in the spent fuel pool (SFP) or in containment, and a spent fuel cask drop accident. Since there are no changes to plant equipment, nor any changes in how fuel is moved, there are no changes to the probability of a fuel handling accident in the spent fuel pool or containment.

Since there are no changes to plant equipment, nor any changes in how a shielded cask would be moved, there are no changes to the probability of a spent fuel cask drop accident.

The consequences of a fuel drop accident in either containment or the spent fuel pool are not affected, since none of the inputs to these fuel drop accidents is affected. There are no physical hardware changes made to the plant. The limiting fuel burnup is not changed, nor is there any change in the source term of radioactivity present in the fuel. Allowing fuel to be stored in the 40 Region B locations currently empty, does not alter the existing FSAR conclusion that a dropped fuel assembly or consolidated storage box could not strike more than one fuel assembly in the storage rack. This is still true since the fuel stored in these 40 locations is stored at the same elevation as fuel in any other storage locations. The FSAR states that the worst fuel handling incident that could occur in the SFP is the drop of a fuel assembly to the pool floor, with resultant failure of 14 fuel rods when the assembly rotates and impacts a protruding structure. Radiological consequences for both the failure of 14 rods and the entire fuel assembly are presented in the FSAR. The storage of fuel in the 40 currently blocked locations does not affect this FSAR sequence of events for the dropped fuel assembly in the SFP accident. The amount of soluble boron concentration necessary in the SFP to ensure that K_{eff} is maintained ≤ 0.95 on a 95/95 bases is increased from 800 ppm to 1400 ppm. However, this increase in required SFP soluble boron concentration does not increase any dose consequences from the fuel drop accident in the SFP. The increase in soluble boron concentration from 800 ppm to 1400 ppm is a result of crediting an additional 600 ppm of SFP soluble boron under normal conditions.

The consequences of a spent fuel cask drop accident in the SFP is not affected, since none of the inputs to the spent fuel cask drop accident is affected. There are no physical hardware changes made to the plant. The limiting fuel burnup is not changed, nor is there any change in the source term of radioactivity present in the fuel. The amount of soluble boron concentration necessary in the SFP to ensure that K_{eff} is maintained ≤ 0.95 on a 95/95 bases is increased from 800 ppm to 1400 ppm. However, this increase in required SFP soluble boron concentration does not increase any dose consequences from the spent fuel cask drop accident in the SFP. The increase in soluble boron concentration from 800 ppm to 1400 ppm is a result of crediting an additional 600 ppm of SFP soluble boron under normal conditions.

With regard to the proposed change in the design features section of Technical Specifications (TS), which would allow higher enrichments in the new fuel storage (dry) vault, there are no FSAR Chapter 14 accident conditions currently analyzed, therefore there can be no change in probability or consequences of an existing accident.

With regard to the proposed change in the design features section of TS, which would allow higher enrichments in the reactor core, enrichment by itself is not a parameter which will affect the probability or consequences of an accident previously analyzed. The effects of enrichment on other reactor core parameters such as shutdown margin, MTC [moderator temperature coefficient] and power distributions is considered by meeting the exi[s]ting TS requirements for these parameters. Also, the reactor core radioactive source term is not affected since the exiting design basis analysis bounds use of the proposed enrichment. Therefore, a change in the maximum enrichment limit will not impact any safety analyses because the important inputs to these analyses are protected by Technical Specifications. Since there are no changes to these existing reactor core TS parameter limits, there will be no effect on the probability or consequences of an accident previously analyzed.

Therefore, based on the above analysis, the proposed changes do not involve a significant increase in 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 changes to be made primarily affect nuclear criticality analysis and do not create a new or different kind of accident. Changes in allowed enrichment, boraflex credit, soluble boron credit, and allowing fuel to be stored in 40 additional locations are all impacts to the SFP criticality analysis. The SFP criticality analysis is part of the basic design of the system and is not an accident. The ability to maintain the SFP $K_{eff} \leq 0.95$, as well as within the 10 CFR 50 App. A GDC62 criteria [General Design Criterion (GDC)–62, "Prevention of criticality in fuel storage and handling," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR part 50] of sub-critical have been evaluated. Criticality impacts are more appropriately discussed under the margin of safety criterion.

Since there are no changes to the plant equipment, there is no possibility of a new or different kind of accident being initiated or affected by equipment issues. There are no changes in how fuel is moved or qualified for storage, so a new accident cannot be initiated from fuel handling related procedures.

Higher SFP soluble boron concentrations are required than previously required to compensate for the positive reactivity insertions from postulated accident conditions (i.e., dropped cask). However, merely increasing the amount of SFP soluble boron required for compensating for the existing analyzed accident does not create the potential for a new or different kind of accident.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the new fuel storage (dry) vault, no new or different kind of accident conditions are created. The existing new fuel storage analysis previously submitted to the NRC is not altered, and already bounds enrichments up to 5.0 w/o U–235.

With regard to the proposed change in the design features section of TS, which would allow higher enrichments in the reactor core, the higher enrichment fuel in the reactor core does not require any new or different plant equipment, and does not change the manner in which currently installed equipment is operated. There are no changes to normal core operation, and the unit will meet all applicable design criteria and will operate within the existing reactor core TS limits. No new failure modes have been created for any system, component or piece of equipment, and no new single failure mechanisms are introduced. Therefore, allowing higher enrichments in the reactor core will not create a new or different kind of accident condition.

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

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

The margin of safety relevant to the SFP are:

- To ensure that the SFP K_{eff} remains ≤ 0.95 on a 95/95 basis to ensure the criticality safety of the SFP.
- To ensure that the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact.

A criticality analysis has been performed to ensure that the spent fuel pool K_{eff} remains ≤ 0.95 on a 95/95 basis under all normal and postulated accident conditions. Thus the margin of criticality safety is not changed. Most of the changes in the criticality analysis are of an input nature, such as a change in allowed enrichment. The only change in methodology is the crediting of soluble boron for normal conditions. The approach used is consistent with WCAP–14416–NP–A. The NRC has previously approved for other plants similar applications for soluble boron credit for normal conditions. The criticality analysis has been performed to ensure that the spent fuel pool K_{eff} remains less than 1.00 on a 95/95 basis even with 0 ppm soluble boron concentration in the SFP. This ensures compliance with GDC62.

The only change that could affect the SFP cooling analysis is allowing 40 additional fuel assemblies to be stored in the SFP. The current design basis heat load analysis already bounds the storage of these fuel assemblies. This ensures that the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact. The current design basis heat load analysis bounds the increased fuel storage.

With regard to the proposed change in the design section of TS, which would allow higher enrichments in the new fuel storage (dry) vault, there is no significant reduction in the margin of safety. The existing new fuel storage analysis previously submitted and approved by the NRC is not altered, and already bounds enrichments up to 5.0 w/o U–235, to ensure that K_{eff} of the new fuel storage racks is maintained ≤ 0.95.

With regard to the proposed change in the design features section of TS, which would allow higher enrichments in the reactor core, enrichment by itself is not a parameter which will affect the margin of safety. The margins of safety, such as fuel DNB protection, fuel melt protection and RCS boundary protection, are met by complying with the safety analysis and associated TS limits. The effects of enrichment on other reactor core parameters such as shutdown margin, MTC and power distributions is considered by meeting the existing TS requirements for these parameters. Therefore, a change in the maximum core enrichment limit will not impact any margins of safety because the important inputs to the safety analyses are protected by Technical Specifications. Since there are no change[s] to these existing reactor core TS parameter limits, there will be no effect on the margin of safety.

Therefore, based on the above analysis, the proposed changes do 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: Lillian M. Cuoco, Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, CT 06385.

NRC Section Chief: James W. Clifford.

Duke Energy Corporation, et al., Docket Nos. 50–413 and 50–414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: December 20, 2001.

Description of amendment request: The amendments would revise the Technical Specifications (TS) for Catawba Nuclear Station, Units 1 and 2, based on a revised radiological dose consequence analysis of a postulated fuel handling accident and weir gate drop accident. The licensee has requested these amendments in accordance with the requirements of 10 CFR 50.67 which addresses the use of an alternate source term at operating reactors, and relevant guidance provided in Regulatory Guide (RG) 1.183.

Specifically, the proposed changes would revise TS 3.7.10, Control Room Area Ventilation System (CRAVS), to require immediate suspension of movement of irradiated fuel with less than two trains of the CRAVS operable. This change is being requested to correct a non-conservatism in this TS.

The proposed change to TS 3.7.11, Control Room Area Chilled Water System, would delete the applicability of the specification during core alterations and during movement of irradiated fuel. This system is not credited as a mitigation system for the postulated fuel handling accident or weir gate drop accident.

The proposed change to TS 3.7.13, Fuel Handling Ventilation Exhaust System, would change the Limiting Condition for Operation to require two trains be operable during the movement of recently irradiated fuel in the fuel building, and to require that movement of recently irradiated fuel in the fuel building be suspended if one train becomes inoperable. Recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. Operability of the Fuel Handling Ventilation Exhaust System would only be required during movement of recently irradiated fuel assemblies. This change is being requested to incorporate the concept of recently irradiated fuel and to correct a non-conservatism in this TS.

The proposed change to TS 3.9.3, Containment Penetrations, would amend the applicability of this specification. Current TS requirements regarding closure of the containment equipment hatch, the personnel airlock and containment penetrations would only apply during movement of recently irradiated fuel assemblies. The applicability of this specification during core alterations would be deleted.

The licensee is requesting these amendments to provide flexibility in scheduling outage tasks and to modify unnecessarily restrictive containment closure and fuel handling building ventilation system requirements. The revised analyses also incorporate updated atmospheric dispersion factors for the Control Room intake pathway.

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

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? No.

An alternate source term calculation has been performed for Catawba Nuclear Station that demonstrates that offsite dose consequences of a postulated fuel handling accident or weir gate drop accident remain within the limits provided sufficient decay has occurred prior to the movement of irradiated fuel without taking credit for certain mitigation features such as ventilation filter systems and containment closure. Irradiated fuel that has not undergone the required decay period of 72 hours is defined to be recently irradiated fuel and the currently approved Technical Specification requirements are applicable when this recently irradiated fuel is being handled.

The proposed amendment would allow core alterations and movement of sufficiently decayed irradiated fuel within the containment building with the equipment hatch, personnel air locks and containment penetrations open. Operation of the

Containment Purge Exhaust System (CPES) is not required during movement of sufficiently decayed fuel. The amendment also would allow movement of irradiated fuel assemblies within the fuel building without the Fuel Handling Ventilation Exhaust System (FHVES) in operation. Movement of the weir gate is permitted without the FHVES in operation provided the irradiated fuel that could be impacted by a drop of the weir gate has undergone a minimum decay period of 19.5 days.

This amendment does not alter the methodology or equipment used directly in fuel handling operations and weir gate movement. Neither ventilation filter systems, the CPES nor the FHVES, is used to actually handle fuel. Neither of these systems is an ''accident initiator'' either in this sense or any other sense. Similarly, neither the equipment hatch, the personnel air locks, nor any other containment penetration, nor any component thereof is an accident initiator.

Actual fuel handling operations and weir gate movement themselves are not affected by the proposed changes. Therefore, the probability of a Fuel Handling Accident and Weir Gate Drop is not affected with the proposed amendment. No other accident initiator is affected by the proposed changes.

For the reasons above, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated.

The Fuel Handling Accident in Containment has been analyzed without credit for filtration by the CPES. Likewise, the Fuel Handling Accident in the Fuel Building and the Weir Gate Drop has been analyzed without credit for filtration by the FHVES. The analyses of these design basis events were conducted with the Alternative Source Term Methodology in accordance with 10 CFR 50.67 and Regulatory Guide 1.183. These analyses show that the resultant radiation doses are within the limits specified in 10 CFR 50.67 and R.G. 1.183.

The TEDE [total effective dose equivalent] radiation doses from the analyses supporting this LAR [license amendment request] have been compared to equivalent TEDE radiation doses estimated with the guidelines of R.G. 1.183 Footnote 7. The new values are shown to be comparable to the results of the previous analyses.

For the reasons above, the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

Does operation of the proposed facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? No.

The proposed change does not involve addition or modification to any plant system, structure, or component. The proposed amendment would increase the time during which the equipment hatch and personnel air locks could be open during core alterations and movement of irradiated fuel. The proposed amendment does not involve any change in the operation of these containment penetrations. Having these penetrations open does not create the possibility of a new accident.

The proposed amendment also would remove the requirements for operability of the CPES and FHVES during core alterations or movement of sufficiently decayed irradiated fuel. It does not alter the operation of these systems beyond their functional capabilities. Modification of the requirements of operability for these systems from the plant Technical Specifications does not create the possibility of a new accident.

The requirements for CRAVS are being revised to immediately suspend movement of irradiated fuel if one CRAVS train becomes inoperable. This change does not have the potential to cause a new or different type of accident.

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

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety? No.

The assumptions and input used in the analysis are conservative as noted below. The design basis Fuel Handling Accidents and Weir Gate Drop have been defined to identify conservative conditions (concerning offsite power and single failure). The source term and radioactivity releases have been calculated pursuant to Regulatory Guide 1.183 and with conservative assumptions concerning prior reactor operation. The control room atmospheric dispersion factors have been calculated with conservative assumptions associated with the release. The conservative assumptions and input noted above ensure that the radiation doses cited in this License Amendment Request are the upper bound to radiological consequences of a Fuel Handling Accident either in Containment or the Fuel Building and the Weir Gate Drop. The analyses show that there is a significant margin between the TEDE radiation doses calculated for the postulated Fuel Handling Accident and the Weir Gate Drop accident using the Alternative Source Term and the acceptance limits of 10 CFR 50.67 and Regulatory Guide 1.183.

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

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

Attorney for licensee: Ms. Lisa F. Vaughn , Legal Department (PB05E), Duke Energy Corporation, 422 South Church Street, Charlotte, North Carolina 28201–1006.

NRC Section Chief: Richard J. Laufer, Acting.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: January 31, 2002.

Description of amendment request: The proposed amendment revises Technical Specification 5.6.5, ''Core Operating Limits Report (COLR),'' to include an additional reference to Entergy Operations, Inc. (Entergy) Topical Report ENEAD–01–P, ''Qualification of Reactor Physics Methods for Pressurized Water Reactors of the Entergy System.'' This topical report documents a Nuclear Regulatory Commission (NRC)-approved methodology that can be utilized to determine core operating limits.

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 add the Entergy Topical Report ENEAD–01–P, ''Qualification of Reactor Physics Methods for Pressurized Water Reactors of the Entergy System,'' to the Core Operating Limits Report (COLR) references is administrative in nature. The topical report has been reviewed and approved by the NRC in [a] Safety Evaluation Report dated September 29, 1995 (0CNA099519). The physical design or operation of the plant is not impacted by this proposed change. The proposed change does not adversely impact transient analysis assumptions or results. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated under the provisions of 10 CFR 50.59.

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.

Adding a reference in the technical specifications to the NRC-approved methodology in Entergy Topical Report ENEAD–01–P is administrative in nature. No physical alterations of plant configuration, changes to the plant operating procedures, or operating parameters are proposed. No new equipment is being introduced, and no equipment is being operated in a manner inconsistent with its design. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies. A 10 CFR 50.59 review will continue to be performed to evaluate specific reload 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 change to reference Entergy Topical Report ENEAD–01–P is administrative in nature. Existing technical specification operability and surveillance requirements are not reduced by the proposed change. The cycle-specific COLR limits for future reloads will continue to be developed based on NRC-approved methodologies and their corresponding physics parameter uncertainties. Technical specifications will continue to require that the core be operated within these limits and specify appropriate actions to be taken if the limits are violated. The COLR-related safety analyses will continue to be performed utilizing NRC-approved methodologies, and specific reload changes will be evaluated per 10 CFR 50.59.

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

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

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

NRC Section Chief: Robert A. Gramm.

Entergy Operations, Inc., Docket No. 50–313, Arkansas Nuclear One, Unit No. 1, Pope County, Arkansas

Date of amendment request: January 31, 2002.

Description of amendment request: The proposed amendment changes administrative Technical Specification 5.5.16 regarding Containment Integrated Leak Rate Testing (ILRT). The change clarifies the statement that the ILRT Program is in accordance with Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program,'' by noting an exception based on Nuclear Energy Institute 94–01, ''Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J.'' The effect of this change will be to allow a one-time extension of the interval (to 15 years) for performance of the next ILRT.

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.

[Appendix J [of] 10 CFR [part] 50] was amended to incorporate provisions for performance-based testing in 1995. The proposed amendment to Technical Specification (TS) 5.5.16 adds a one-time extension to the current interval for Type A testing (i.e., the integrated leak rate test). 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. In addition, at ANO–1 [Arkansas Nuclear One, Unit 1,] 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 leaktightness.

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 test itself is not being modified, but is only intended to be 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. Further, the extended test interval would have a minimal effect on this risk since Type B and C testing detect 97% of potential leakage paths. For the requested change in the ANO–1 ILRT interval, it was determined that the risk contribution of leakage will increase 0.19%. 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., Washington, DC 20005–3502.

NRC Section Chief: Robert A. Gramm.

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

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

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

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

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

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

Date of amendment request: November 30, 2001.

Description of amendment request: A change is proposed to Surveillance Requirement (SR) 3.0.3 to allow a longer period of time to perform a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''. . . up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement would be added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

The Nuclear Regulatory Commission (NRC) staff issued a notice of opportunity for comment in the **Federal Register** on June 14, 2001, (66 FR 32400), on possible amendments concerning missed surveillances, 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 September 28, 2001, (66 FR 49714). The licensees affirmed the applicability of the following NSHC determination in its application dated November 30, 2001.

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

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

The proposed change relaxes the time allowed to perform a missed surveillance. The time between surveillances is not an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Any reduction in confidence that a standby system might fail to perform its safety function due to a missed surveillance is small and would not, in the absence of other unrelated failures, lead to an increase in consequences beyond those estimated by existing analyses. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 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. A missed surveillance will not, in and of itself, introduce new failure modes or effects and any increased chance that a standby system might fail to perform its safety function due to a missed surveillance would not, in the absence of other unrelated failures, lead to an accident beyond those previously evaluated. The addition of a requirement to assess and manage the risk introduced by the missed surveillance will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The extended time allowed to perform a missed surveillance does not result in a significant reduction in the margin of safety. As supported by the historical data, the likely outcome of any surveillance is verification that the LCO [Limiting Condition for Operation] is met. Failure to perform a surveillance within the prescribed frequency does not cause equipment to become inoperable. The only effect of the additional time allowed to perform a missed surveillance on the margin of safety is the extension of the time until inoperable equipment is discovered to be inoperable by the missed surveillance. However, given the rare occurrence of inoperable equipment, and the rare occurrence of a missed surveillance, a missed surveillance on inoperable equipment would be very unlikely. This must be balanced against the real risk of manipulating the plant equipment or condition to perform the missed surveillance. In addition, parallel trains and alternate equipment are typically available to perform the safety function of the equipment not tested. Thus, there is confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chiefs: Anthony J. Mendiola and James W. Clifford.

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

Date of amendment request: January 18, 2002.

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for St. Lucie Units 1 and 2 to remove the numerical working hour limits stated in the TS. Site personnel working hours currently are and will continue to be controlled by administrative procedures. The change is consistent with Technical Specifications Task Force (TSTF) Item TSTF–258, Rev. 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) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments are administrative in nature and they do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. These proposed changes do not change the existing administrative controls on plant staff working hours. Any future changes to these procedures will be controlled under established procedure control processes that will ensure the administrative controls on work hours remain effective. Further, the proposed changes do not alter the design, function, or operation of any plant component. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The changes being proposed are administrative in nature and do not introduce a new mode of plant operation or surveillance requirement, nor involve a physical modification to the plant. Therefore, the design, function, or operation of any plant component is not altered. The changes propose to relocate specific controls for plant staff working hours from the TS to existing administrative procedures. The specific controls for plant staff working hours are described in these procedures and require a deliberate decision-making process to manage the potential for impaired personnel performance. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes conform closely to the industry and NRC approved TSTF–258 Rev. 4 and relate to the relocation of TS specific working hour limits and controls to administrative procedures that control working hours. The specific controls for working hours of reactor plant staff are described in procedures that require a deliberate decision-making process to manage the potential for impaired personnel performance. Furthermore, any future changes to these procedures will be controlled under established procedure control processes that will ensure the administrative controls on work hours remain effective. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: January 25, 2002.

Description of amendment request: The proposed amendments would revise the St. Lucie Plant, Units 1 and 2, Technical Specifications, Appendix B, ''Environmental Protection Plan (Non-Radiological)'' to incorporate the revised terms and conditions of the Incidental Take Statement included in the Biological Opinion issued by the National Marine Fisheries Service on May 4, 2001, as clarified by NMFS letter dated October 8, 2001. These amendments also incorporate administrative revisions necessary to change references to the National Pollutant Discharge Elimination System Permit to the Wastewater Permit, based on a change in administrative authority over these permits.

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

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

The changes are administrative in nature and would in no way affect the initial conditions, assumptions, or conclusions of the St. Lucie Unit 1 or Unit 2 accident analyses. In addition, the proposed changes would not affect the operation or performance of any equipment assumed in the accident analyses. Based on the above information, we conclude that the proposed changes would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The changes are administrative in nature and would in no way impact or alter the configuration or operation of the facilities and would create no new modes of operation. We conclude that the proposed changes would not create the possibility of a new or different kind of accident.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

The changes are administrative in nature and would in no way affect plant or equipment operation or the accident analysis. We conclude that the proposed changes would not result in a significant reduction in a margin of safety.

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

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: January 25, 2002.

Description of amendment request: The proposed amendment would revise Section 4.8.1.1.2.g.2 of the St. Lucie Units 1 and 2 Technical Specifications (TS), which currently requires a test of the diesel fuel oil system piping at elevated pressure once every 10 years. In lieu of hydrostatic testing, the diesel fuel oil systems will be included in the population of systems subjected to periodic system pressure testing at normal operating conditions required by the American Society of Mechanical Engineers Code for Class 3 systems in accordance with the inservice inspection program.

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

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

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because industry experience has shown that an inservice leak test conducted at normal operating temperature and pressure is just as effective at finding leakage as a hydrostatic test conducted at 110 percent of the design pressure. Therefore, there is no increase in the probability or consequences of previously evaluated accidents. Also, note that the diesel generator fuel oil system is not specifically modeled in the St. Lucie probability safety assessment (PSA). Based on the St. Lucie PSA, the diesel generator failure probability is dominated by failure modes other than fuel oil pipe rupture. The total diesel generator failure probability is on the order of 1E–2, with the contribution from fuel oil pipe rupture on the order of 1E–5 (i.e., three orders of magnitude below the EDG [emergency diesel generator] failure probability).

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

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments provide an alternative method of leak detection for the required 10-year inservice inspection. They do not result in an operational condition different from that which has already been considered by TS. Therefore, the changes do not create the possibility of a new or different kind of accident or malfunction.

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

The alternative method of leak detection has no impact on the consequences of any analyzed accident and does not significantly change the failure probability of equipment that provides protection for the health and safety of the public. Therefore, there is no significant decrease 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 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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

NRC Section Chief: Richard P. Correia.

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

Date of amendment request: January 14, 2002.

Description of amendment request: The proposed amendment would revise the Kewaunee Nuclear Power Plant Technical Specifications (TS) 3.10.f, ''Inoperable Rod Position Indicator Channels,'' to provide an allowed outage time (AOT) for the Individual Rod Position Indicator (IRPI) system of 24 hours with more than one IRPI per group inoperable. The TS did not previously have an explicit AOT for this condition. In addition, the proposed amendment would reformat TS 3.10.f using MicroSoft Word to more closely resemble the format of Improved Standard Technical Specification (ISTS) to improve clarity.

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

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

The format changes are administrative in nature and therefore have no effect on the probability or consequences of an accident. The Individual Rod Position Indicator (IRPI) System is not an accident initiator. Therefore, any change to the system would not effect the probability of an accident previously evaluated. The risk of core damage/release of radioactivity would not increase with the other reactor condition monitors still functional along with the plant mode remaining the same.

The proposed changes provide more time to troubleshoot and restore the system, which would keep the reactor in a steady state condition, rather than to challenge the plant with a reduction in power. The addition of hourly reactor temperature checks as well as placing the rod controls to manual are added to temporarily increase the surveillance on the reactor due to loss of the IRPI system during the IRPI AOT. Since IRPI's are not an accident initiator and compensatory measures have been added to ensure rod position is known incase one or more IRPIs are inoperable, this amendment does not involve an increase in 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 format changes are administrative in nature and therefore have no effect on the probability or consequences of a new or different kind of accident from any accident previously evaluated. The primary function of the IRPI system is to monitor the position of each rod and send that information to the control room. A failure of this system will not result in an accident.

The proposed changes do not involve a change to the physical plant or operations. Operations currently monitors power tilt, excore detectors, thermocouples, and rod movement when IRPIs become inoperable. The extra surveillance requirements added by the ISTS for the AOT are there to cover the loss of information when the IRPIs are OOS. The change to 24 hours for troubleshooting when more than one IRPI channel per group is inoperable would therefore not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The format changes are administrative in nature and therefore are not involved in a significant reduction in the margin of safety. Margin of safety relates to actual rod position in relation to each other. This margin is controlled by rod misalignment requirements. The IRPIs provide indication of that position which the operators have other means of determining should the need arise. The implementation of this proposed amendment ensures continued close monitoring of rod position but also adds hourly documentation of the reactor coolant temperature requirement as well. The proposed change provides more time to troubleshoot and restore the system, which would keep the reactor in a steady state condition, rather than to challenge the plant with a reduction in power. Therefore, NMC concludes that there is not 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: Bradley D. Jackson, Esq., Foley and Lardner, P.O. Box 1497, Madison, WI 53701–1497.

NRC Section Chief: William D. Reckley, Acting Section Chief.

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

Date of amendment request: January 28, 2002.

Description of amendment request: The proposed amendment would revise the Core Operating Limits Report (COLR) analytical methods referenced in Technical Specification (TS) 5.6.5.b. Specifically, the changes would add references to two NRC-approved Framatome ANP, Inc., reports: (1) EMF–2310(P)(A), Revision 0, ''SRP [Standard Review Plan] Chapter 15 Non-LOCA [loss-of-coolant accident] Methodology for Pressurized Water Reactors [PWRs],'' dated May 2001, and (2) EMF–2328(P)(A), Revision 0, ''PWR Small Break LOCA Evaluation Model, S–RELAP5 Based,'' dated March 2001. Existing references in TS 5.6.5.b describing Exxon Nuclear Company's large-break LOCA evaluation model would be deleted.

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:

Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, ''Issuance of Amendment.'' The following evaluation supports the finding that operation of the facility in accordance with the proposed change would not:

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

The proposed license amendment removes a safety analysis methodology and adds two new safety analysis methodologies in TS 5.6.5.b. Accidents previously evaluated will be unaffected because they will continue to be analyzed using applicable methodologies approved by the Nuclear Regulatory Commission to ensure all required safety limits are met. The proposed amendment does not affect the acceptance criteria for loss-of-coolant accidents (LOCA) or non loss-of-coolant accidents. As such, the proposed amendment does not increase the probability or consequences of an accident. The proposed amendment does not involve operation of the required structures, systems or components (SSCs) in a manner or configuration different from those previously recognized or evaluated.

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

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

The proposed amendment does not involve a physical alteration of any SSC or a change in the way any SSC is operated. The proposed amendment does not involve operation of any required SSCs in a manner or configuration different from those previously recognized or evaluated. No new failure mechanisms will be introduced by changes being requested.

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

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

The proposed amendment does not, by itself, introduce a failure mechanism. The proposed amendment does not involve any physical changes to the plant or manner in which the plant was operated. The proposed changes do not affect the acceptance criteria for loss-of-coolant or non-loss-of-coolant accidents. All required safety limits will continue to be analyzed using methodologies approved by the Nuclear Regulatory Commission.

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

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

Attorney for licensee: Arunas T. Udrys, Esquire, Consumers Energy Company, 212 West Michigan Avenue, Jackson, Michigan 49201.

NRC Section Chief: William D. Reckley (Acting).

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

Date of amendment request: January 4, 2002.

Description of amendment request: The proposed amendment would add Technical Specification (TS) 3/4.3.10, ''Mechanical Vacuum Pump Trip Instrumentation.''

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

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

No. The proposed amendment would add a new TS section for the instrumentation that provides the automatic tripping of the mechanical vacuum pumps when high radiation is detected in the main steamlines. The proposed change has no effect on any structures, systems, or components (SSCs) since the mechanical vacuum pump automatic trip function is already part of the existing plant design. The new TS requirements are being added because automatic tripping of the mechanical vacuum pumps is credited for mitigating the radiological consequences of a control rod drop accident (CRDA), and, as such, a TS LCO must be established to meet the requirements stated in 10 CFR 50.36(c)(2)(ii), Criterion 3. Since the proposed change only establishes TS requirements for an existing function, and there are no effects to any SSCs, there is no impact on the CRDA analysis. 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?

No. The proposed amendment does not change the design function or operation of any SSCs. Plant operation will not be affected by the proposed change and no new failure mechanisms, malfunctions, or accident initiators will be created. 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?

No. The proposed amendment only establishes TS requirements for an existing function and will not change any plant operating parameters. The licensee's submittal stated that the proposed change does not affect the radiological consequences for a CRDA. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment requests: March 21, 2001 as superseded by letter dated October 24, 2001.

Description of amendment requests: The proposed amendments would revise Technical Specification (TS) 5.5.2.12 pertaining to ventilation filter testing program (VFTP). Specifically, the proposed amendments would: (a) in TS 5.5.2.12 lead paragraph, and TS 5.5.2.12d, delete reference to Regulatory Guide (RG) 1.52 and American Society of Mechanical Engineers (ASME) N510–1989; the testing frequency will continue to be in accordance with RG 1.52, Revision 2; (b) in TS 5.5.2.12a and TS 5.5.2.12b, refer to American National Standards Institute (ANSI) Code N510–1975 instead of ASME N510–1989, and add a note to provide clarification regarding high energy particulate air (HEPA) filter qualification and testing methodology; and (c) in TS 5.5.2.12c, include specific temperature and relative humidity for laboratory testing of charcoal adsorber samples.

The October 24, 2001, submittal supercedes in its entirety the licensee's March 21, 2001, submittal which was previously noticed in the **Federal Register** on April 18, 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) Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change is to change the reference to ASME Code in subsection 5.5.2.12.a and 5.5.2.12.b from ASME N510–1989 to ANSI N510–1975. Technical Specification (TS) 3.7.11, ''Control Room Emergency Air Cleanup System'' (CREACUS), Surveillance Requirement (SR) 3.7.11.2 and TS 3.7.14, ''Fuel Handling Building Post-Accident Cleanup Filter System'' (PACU), SR 3.7.14.2 requires CREACUS and PACU filter testing in accordance with the Ventilation Filter Testing Program (VFTP).

SONGS [San Onofre Nuclear Generating Station] TS 5.5.12.a, ''Ventilation Filter Testing Program,'' states that the in-place HEPA filter testing is performed in accordance with RG 1.52, Revision 2 and ASME N510–1989. The discrepancy arises because the HEPA filter testing method used at SONGS does not entirely meet the methodology which are delineated in ASME N510–1989. In particular, the CREACUS in-place HEPA filter testing uses a method (Alternate Shroud Test) which is no longer specified in ASME N510–1989. But this method is specified in ANSI N510–1975 and was used when the plant was licensed. In addition, the PACU in-place HEPA filter testing methodology which is employed at SONGS, has a downstream point location which differs from the location suggested in ASME N510–1989.

ANSI N510–1975, while providing a suggestion where downstream sample could be located, nevertheless does not provide a specific location. The test acceptance criteria are the same for methods cited in ANSI N510–1975 and ASME N510–1989. The method which is employed at SONGS provides more conservative results because the test is performed on individual HEPA filters, which ensures that each of the HEPA filters in the tested bank meets the acceptance criteria, as compared to the method suggested in ASME N510–1989.

The locations of the PACU HEPA downstream sample points are different from the locations suggested in ASME N510–1989, though they meet the requirements delineated in ANSI N510–1975. ANSI N510–1975 requires that a single representative downstream sample point be established, if possible, at the location where adequate mixing may be achieved, or at a point downstream of a fan, or multiple downstream sampling points may be used (such as in the Alternate Shroud Technique used in the CREACUS system) if a single downstream sample point is not feasible.

Since the HEPA filters are tested to the same acceptance criteria, and the testing methodology is permitted by ANSI N510–1975, to which the plant was licensed, it is concluded, that the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Section 5.5.2.12.c will be modified by specifying the temperature and relative humidity for laboratory testing of charcoal adsorber samples. This modification clarifies that samples shall be obtained in accordance with RG 1.52, Revision 2, and tested per methodology of ASTM D3803–1989, at 30'C and relative humidity of 70%. This clarification eliminates possible misinterpretation of the current wording.

The proposed change will also include Note 1 which clarifies the inplace testing of charcoal adsorbers and HEPA filters. Based on the provisions of section 10.4 of ASME N510–1989, this Note allows replacement of DOP [dioctyl phthalate] with a suitable alternate.

The proposed change also clarifies the statement of subsection 5.5.2.12.d. Pressure drop testing across combined HEPA filters, the prefilters, and the charcoal adsorbers is industry-wide practice which is based on good engineering practice and operating experience. This change will not increase the probability or consequences of an accident previously evaluated.

Therefore, the probability or consequences of an accident previously evaluated will not be increased by operating the facility in accordance with this proposed change.

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

Response: No.

The proposed change does not change the design or configuration of the plant. The proposed change is to change the reference to ASME Code in subsection 5.5.2.12.a and 5.5.2.12.b from ASME N510–1989 to ANSI N510–1975 to reflect the standard used. Section 5.5.2.12.c will be modified by specifying the temperature and relative humidity for laboratory testing of charcoal adsorber samples. This is done for clarification purposes. Also, subsection 5.5.2.12.d will be changed by deleting the references to RG 1.52, Revision 2, and ASME N510–1989 regarding pressure drop test across HEPA filters. RG 1.52, Revision 2 and ASME N510–1989 do not require pressure drop test[ing] across HEPA filters.

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

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

Response: No.

The proposed change is to change the reference to ASME Code in subsections 5.5.2.12.a, and 5.5.2.12.b from ASME N510–1989 to ASME N510–1975. The CREACUS units HEPA filters are currently tested to N510–1975. Although the test methodology is slightly different than that in N510–1989, the acceptance criteria are the same and the current methodology is conservative. Thus the current testing satisfies the acceptance criteria of N510–1989, even though the test method is different. Section 5.5.2.12.c will be clarified by specifying the temperature and relative humidity for laboratory testing of charcoal adsorber samples.

The current methodology for HEPA filter testing will not change as a result of the proposed change. Also, deletion of references to RG 1.52, Revision 2 and ASME N510–1989 from subsection 5.5.2.12.d clarifies this section because these standards do not require HEPA filters pressure drop test. Consequently, there is no change to the design or operation of the plant as a result of this change.

Therefore, the operation of the facility in accordance with this proposed change will not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Douglas K. Porter, Esquire, Southern California Edison Company, 2244 Walnut Grove Avenue, Rosemead, California 91770.

NRC Section Chief: Stephen Dembek.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/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.*

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

Date of application for amendment: April 4, 2001, as supplemented on October 12, November 28, November 30, December 7, and December 20, 2001.

Brief description of amendment: The amendment deletes Technical Specifications (TSs) 5.3.1.B and 5.3.1.C. These TSs restricted the handling of heavy loads over irradiated fuel stored in the spent fuel pool. The basis for deleting these TSs is the upgrade of the reactor building crane and associated handling systems to a single-failure proof system.

Date of Issuance: January 23, 2002.

Effective date: January 23, 2002, and shall be implemented within 60 days of issuance.

Amendment No.: 223.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 31702).

The supplemental letters dated October 12, November 28, November 30, December 7, and December 20, 2001, provided clarifying information within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 23, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 9, 2001.

Brief description of amendment: The amendment incorporated changes to the Technical Specifications (TSs) to provide consistency with the changes to 10 CFR 50.59, ''Changes tests, and experiments,'' as published in the **Federal Register** (FR) (64 FR 53582), dated October 4, 1999. Specifically, the changes replace the term ''safety evaluation'' with ''10 CFR 50.59 evaluation'' and ''unreviewed safety question'' with ''requires NRC approval pursuant to 10 CFR 50.59.''

Date of Issuance: January 28, 2002.

Effective date: January 28, 2002, and shall be implemented within 60 days of issuance.

Amendment No.: 224.

Facility Operating License No. DPR–16: Amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44162).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated January 28, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: April 4, 2001.

Brief description of amendments: The amendments revise Technical Specifications (TSs) 3.3.12 and 3.9.2 and associated bases pages to (1) clarify operability requirements for the boron dilution alarm system (BDAS) by adding Mode 6 applicability to TS 3.3.12, (2) ensure appropriate operator action when the BDAS is declared inoperable by adding a note to TS 3.9.2 and (3) delete Action 3.9.2.B.2.

Date of issuance: January 29, 2002.

Effective date: January 29, 2002, and shall be implemented within 45 days of the date of issuance.

Amendment Nos.: Unit 1—138, Unit 2—138, Unit 3—138.

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

Date of initial notice in **Federal Register:** May 2, 2001 (66 FR 22024). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: May 7, 2001, as supplemented June 29, 2001.

Brief description of amendment: The amendment revises Technical Specification (TS) 3.4.3 and the associated Surveillance Requirement (SR) to eliminate the pressurizer water volume value in the specification and change ''volume'' to ''level'' in TS 3.4.3, SR 4.4.3, and the associated Bases.

Date of issuance: February 5, 2002.

Effective date: February 5, 2002.

Amendment No. 109.

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

Date of initial notice in **Federal Register:** July 25, 2001 (66 FR 38760).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: November 11, 2001.

Brief description of amendment: The amendment revises Technical Specification Surveillance Requirement (SR) 3.0.3 to allow a longer period of time before entering a limiting condition for operation in the event of a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: January 25, 2002.

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

Amendment No.: 145.

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

Date of initial notice in **Federal Register:** December 12, 2001 (66 FR 64289).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 25, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: April 26, 2001.

Brief description of amendment: The amendment approves a change to the Technical Specifications (TSs) and Bases related to reactor coolant pump flywheel inspection requirements and reactor coolant system structural integrity. The changes add Section 6.22, ''Reactor Coolant Pump Flywheel Inspection Program'' to the TSs and relocate the requirements of TS 3/4.4.10, ''Reactor Coolant System, Structural Integrity'' to the Technical Requirements Manual.

Date of issuance: February 1, 2002.

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

Amendment No.: 264.

Facility Operating License No. DPR–65: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 30, 2001 (66 FR 29351).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 1, 2002.

No significant hazards consideration comments received: No.

Duke Energy Corporation, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station (MNS), Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: March 22, 2001, as supplemented by letter dated October 11, 2001.

Brief description of amendments: The amendments revised the current MNS Technical Specifications (TS) surveillance requirement (SR) for the methodology and frequency for the chemical analyses of the ice condenser ice bed. Also, these amendments add a new TS SR to address sampling requirements for ice additions to the ice bed. In addition, the amendments revise the current MNS TS acceptance criteria and surveillance frequency for the inspection of ice condenser ice basket flow channel areas.

Date of issuance: February 1, 2002.

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

Amendment Nos.: 201 and 182.

Facility Operating License Nos. NPF–9 and NPF–17: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 11, 2001 (66 FR 36339).

The supplement dated October 11, 2001, provided clarifying information that did not change the scope of the March 22, 2001, application nor the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 1, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: September 20, 2001.

Brief description of amendment: The amendment revised Technical Specification Section 6.8.4.a to delete the requirements to have a program to obtain and analyze samples of reactor coolant and containment atmosphere under accident conditions (post accident sampling system).

Date of issuance: January 30, 2002.

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

Amendment No.: 222.

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

*Date of initial notice in **Federal Register:*** October 31, 2001 (66 FR 55013).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: October 23, 2001.

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.

Date of issuance: February 6, 2002.

Effective date: As of the date of issuance to be implemented within 90 days.

Amendment No.: 210.

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

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64293).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: February 20, 2001, as supplemented by letters dated July 13, 2001, and December 28, 2001.

Brief description of amendments: The amendments change the Technical Specifications (TS) Section 3.8.1, ''A.C. Sources-Operating,'' to extend to 14 days the allowable completion time for the required actions associated with restoration of an inoperable Division 1 or Division 2 Emergency Diesel Generator. In addition, the amendments change the TS completion time period associated with discovery of failure to meet TS limiting condition of operation 3.8.1 from 10 days to 17 days.

Date of issuance: January 30, 2002.

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

Amendment Nos.: 150 and 136.

Facility Operating License Nos. NPF–11 and NPF–18: The amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** March 21, 2001 (66 FR 15925).

The supplemental letter 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 January 30, 2002.

No significant hazards consideration comments received: No.

Exelon Generation Company, Docket No. 50–352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: September 14, 2001.

Brief description of amendment: The amendment revises Technical Specification (TS) Figure 3.4.6.1–1, ''Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure,'' to extend the use of the reactor pressure vessel (RPV) pressure-temperature limit curves for one additional fuel cycle and approves a modification to the TS Table 4.4.6.1.3–1, ''Reactor Vessel Surveillance Program—Withdrawal Schedule,'' RPV surveillance capsule withdrawal schedule.

Date of issuance: January 30, 2002.

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

Amendment No.: 155.

Facility Operating License No. NPF–39: This amendment revised the TS.

*Date of initial notice in **Federal Register:*** November 28, 2001 (66 FR 59506).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 30, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: March 28, 2001, as supplemented May 1, 2001, and June 13, 2001.

Brief description of amendments: These amendments relocate certain Beaver Valley technical specifications (TSs) to the Licensing Requirements Manual and the Offsite Dosage Calculation Manual. The major change proposed in this request involves the application of the TS screening criteria of 10 CFR 50.36.

Date of issuance: January 24, 2002.

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

Amendment Nos.: 246 and 124.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** June 20, 2001 (66 FR 33111).

The May 1 and June 13, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the initial **Federal Register** notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 24, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: October 29, 2001, as supplemented December 17, 2001.

Brief description of amendments: These amendments revised Technical Specification Section 3.9.3 to reduce the minimum decay time required prior to fuel movement from 150 hours to 100 hours.

Date of issuance: January 29, 2002.

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

Amendment Nos.: 247 and 126.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 26, 2001 (66 FR 66465). The December 17, 2001, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the application beyond the scope of the original **Federal Register** notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: January 18, 2001, as supplemented by letters dated February 20, June 9, June 26, June 29, October 31, and December 19, 2001.

Brief description of amendments: The amendments changed the technical specifications associated with modifying the maximum power levels permissible with inoperable main steam safety valves.

Date of issuance: January 29, 2002.

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

Amendment Nos.: 248 and 127.

Facility Operating License Nos. DPR–66 and NPF–73: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** July 27, 2001 (66 FR 39211).

The February 20, June 9, June 26, June 29, October 31, and December 19, 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the amendment beyond the scope of the original notice.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: July 25, 2001.

Brief description of amendment: The amendment approved increases to the BVPS–2 Technical Specification boron concentration limits for the refueling water storage tank, accumulators, and the reactor coolant system/refueling canal during Mode 6.

Date of issuance: January 28, 2002.

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

Amendment No: 125.

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

Date of initial notice in ***Federal Register:*** October 3, 2001 (66 FR 50468).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 28, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendment: February 21, 2001, as supplemented April 26, 2001.

Brief description of amendment: The amendment revised the Improved Technical Specification 3.3.8, to clarify actions to be taken in the event that one or more channels of the loss of voltage or degraded voltage Emergency Diesel Generator start functions become inoperable.

Date of issuance: January 29, 2002.

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

Amendment No.: 202.

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

Date of initial notice in ***Federal Register:*** March 21, 2001 (66 FR 15925). The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 29, 2002.

No significant hazards consideration comments received: No.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant, Units 3 and 4, Miami-Dade County, Florida

Date of application for amendments: October 17, 2001.

Brief description of amendments: The amendments revised Technical Specification 6.8.4.h, to allow a one-time change in the containment integrated leakage rate test interval from the required 10 years to a test interval of 15 years.

Date of issuance: January 29, 2002.

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

Amendment Nos: 218 and 212.

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

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59507).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2002.

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: July 17, 2001.

Brief description of amendments: The amendments revise Technical Specification 4.0.3 and its associated Bases to provide for a delay period in which to perform a surveillance which has been discovered not to have been performed within its specified frequency.

Date of issuance: January 28, 2002.

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

Amendment Nos.: 263 and 245.

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

Date of initial notice in ***Federal Register:*** August 22, 2001 (66 FR 44175).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 28, 2002.

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 7, 2001.

Brief description of amendments: The amendments would create Technical Specification (TS) 3.0.6 and associated bases to allow equipment that was removed from service or declared inoperable to be returned to service under administrative controls solely to perform the testing required to demonstrate its operability or the operability of other equipment.

Date of issuance: February 1, 2002.

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

Amendment Nos.: 264 and 246.

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

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59508).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated February 1, 2002.

No significant hazards consideration comments received: No.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: April 11, 2001.

Brief description of amendment: The proposed change would revise Technical Specifications 4.2, Fuel Storage, and 5.6.5, Spent Fuel Pool Water Chemistry Program, by adding applicability statements that specify that these specifications apply only when irradiated fuel is stored in the spent fuel storage pool. These changes are being made to facilitate dismantlement of the spent fuel storage pool upon removal of the last irradiated fuel assembly from the spent fuel storage pool to the onsite independent spent fuel storage installation (ISFSI).

Date of issuance: February 6, 2002.

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

Amendment No.: 166.

Facility Operating License No. DPR–36: The amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** August 8, 2001 (66 FR 41623).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 6, 2002.

No significant hazards consideration comments received: No.

Nuclear Management Company, LLC, Docket Nos. 50–266 and 50–301, Point Beach Nuclear Plant, Units 1 and 2, Town of Two Creeks, Manitowoc County, Wisconsin

Date of application for amendments: November 19, 2001.

Brief description of amendments: The amendment changes Surveillance Requirement (SR) 3.0.3 to allow a longer period of time before entering a Limiting Condition of Operations in the event of a missed surveillance. The time is extended from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to '' * * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: January 28, 2002.

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

Amendment Nos.: 202 and 207.

Facility Operating License Nos. DPR–24 and DPR–27: Amendments revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** November 28, 2001 (66 FR 59510).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 28, 2002.

No significant hazards consideration comments received: No.

PPL Susquehanna, LLC, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: May 31, 2001, as supplemented December 5, 2001.

Brief description of amendment: The amendment revised the minimum critical power ratio safety limits.

Date of issuance: January 31, 2002.

Effective date: As of date of issuance and shall be implemented upon startup following the Unit 1 twelfth refueling and inspection outage.

Amendment No.: 199.

Facility Operating License No. NPF–14: This amendment revised the Technical Specifications.

Date of initial notice in ***Federal Register:*** September 5, 2001 (66 FR 46480). The supplemental letter provided additional information but did not change the initial no significant hazards consideration determination or expand the amendment beyond the scope of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated January 31, 2002.

No significant hazards consideration comments received: No.

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

Date of application for amendments: November 28, 2000.

Brief description of amendments: The amendments expanded the allowable suppression chamber-to-drywell vacuum breaker setpoint range.

Date of issuance: January 29, 2002.

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

Amendment Nos.: 198 and 173.

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

Date of initial notice in ***Federal Register:*** January 10, 2001 (66 FR 2023).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 29, 2002.

No significant hazards consideration comments received: No.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: May 21, 2001.

Brief description of amendment: The amendment deletes the Definitions, Limiting Conditions for Operation and Surveillance Requirements, and several sections from the Administrative Controls portion of the technical specifications once the spent nuclear fuel has been transferred from the 10 CFR part 50 licensed site to the 10 CFR part 72 licensed Independent Spent Fuel Storage Installation.

Date of issuance: February 5, 2002.

Effective date: February 5, 2002, to be implemented within 30 days after the transfer of the last cask of spent nuclear fuel from the spent fuel pool to the Independent Spent Fuel Storage Installation is complete.

Amendment No.: 129.

Facility Operating License No. DPR–54: The amendment revised the Technical Specifications.

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

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2002.

No significant hazards consideration comments received: No.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: June 7, 2001.

Brief description of amendment: The amendment deletes administrative requirements which are no longer applicable once the spent nuclear fuel has been transferred from the 10 CFR part 50 licensed site to the 10 CFR part 72 licensed Independent Spent Fuel Storage Installation. Other requirements are transferred to the Rancho Seco Quality Manual.

Date of issuance: February 5, 2002.

Effective date: February 5, 2002, to be implemented within 30 days after the transfer of the last cask of spent nuclear fuel from the spent fuel pool to the Independent Spent Fuel Storage Installation is complete.

Amendment No.: 130.

Facility Operating License No. DPR–54: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** July 11, 2001 (66 FR 36344).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2002.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: November 7, 2001.

Brief description of amendment: The amendment revises Surveillance Requirement (SR) 3.0.3 to extend the delay period before entering a limiting condition for operation upon a missed SR from the current limit of ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is less'' to ''* * * up to 24 hours or up to the limit of the specified Frequency, whichever is greater.'' In addition, the following requirement is added to SR 3.0.3: ''A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.''

Date of issuance: February 5, 2002.

Effective date: February 5, 2002, and shall be implemented within 60 days of the date of issuance.

Amendment No.: 147.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64307).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2002.

No significant hazards consideration comments received: No.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: November 7, 2001 (ULNRC–04557).

Brief description of amendment: The amendment revised Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 in the Technical Specifications (TSs) on reactor trip system (RTS) instrumentation. The change to SR 3.3.1.2 replaces the reference to the nuclear instrumentation system (NIS) channel output by a reference to the power range channel output and deletes Note 1 to the SR. The change to SR 3.3.1.3 is editorial.

Date of issuance: February 5, 2002.

Effective date: February 5, 2002, and shall be implemented, including adding the changes to the Bases of the Technical Specifications as described in the licensee's application of November 7, 2001, before the startup from refueling outage 12, which is scheduled for the Fall of 2002.

Amendment No.: 148.

Facility Operating License No. NPF–30: The amendment revised the Technical Specifications.

*Date of initial notice in **Federal Register:*** December 12, 2001 (66 FR 64308).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated February 5, 2002.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50–338, North Anna Power Station, Unit 1, Louisa County, Virginia Date of application for amendment: January 9, 2001.

Brief description of amendment: This amendment revises the Facility Operating License (FOL) and Technical Specifications (TS) to remove obsolete license conditions, make editorial changes in the FOL, relocate license conditions, and remove redundant license conditions covered elsewhere in the license.

Date of issuance: January 31, 2002.

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

Amendment No.: 230.

Facility Operating License No. NPF–4: Amendment changes the FOL and TS.

*Date of initial notice in **Federal Register:*** February 21, 2001 (66 FR 11064).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated January 31, 2002.

No significant hazards consideration comments received: No.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 11th day of February 2002.

John A. Zwolinski,

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

[FR Doc. 02–3750 Filed 2–18–02; 8:45 am]

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OFFICE OF PERSONNEL MANAGEMENT

Excepted Service

AGENCY: Office of Personnel Management.

ACTION: Notice.

SUMMARY: This gives notice of positions placed under Schedule A, B, and C in the excepted service, as required by Civil Service Rule VI, Exceptions from the Competitive Service.

FOR FURTHER INFORMATION CONTACT: Pam Shivery, Director, Washington Service Center, Employment Service (202) 606–1015.

SUPPLEMENTARY INFORMATION: Appearing in the listing below are 1 Schedule A authority and the individual authorities established under Schedule C between January 1, 2002, and January 31, 2002. Future notices will be published on the fourth Tuesday of each month, or as soon as possible thereafter. A consolidated listing of all Excepted Service authorities as of June 30 is published each year.

Schedule A

Social Security Administration

Temporary and time-limited positions in the Ticket to Work and Work Incentives Advisor Panel. No Employees may be appointed after November 17, 2007. Effective November 19, 2001.

Schedule C

The following Schedule C authorities were established during January 2002:

Council on Environmental Quality

Special Assistant to the Chair, Council on Economic Quality. Effective January 11, 2002.

Department of Agriculture

Confidential Assistant to the Administrator, Farm Service Agency. Effective January 14, 2002.

Director, Intergovernmental Affairs to the Assistant Secretary for Congressional Relations. Effective January 17, 2002.

Staff Assistant to the Under Secretary for Marketing and Regulatory Programs. Effective January 24, 2002.

Confidential Assistant to the Administrator, Food and Nutrition Service. Effective January 25, 2002.

Confidential Assistant to the Assistant Secretary for Congressional Relations. Effective January 30, 2002.

Special Assistant to the Chief, Natural Resource Manager. Effective January 30, 2002.