request is by a person other than the Licensee. Because of potential disruptions in delivery of mail to United States Government offices, it is requested that answers and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to (301) 415–1101 or by e-mail to *hearingdocket@nrc.gov*, and also to the Office of the General Counsel either by means of facsimile transmission to (301) 415–3725 or by e-mail to *OGCMailCenter@nrc.gov*. If a person other than the Licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.714(d).

If a hearing is requested by the Licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

Pursuant to 10 CFR 2.202(c)(2)(i), the Licensee may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section III above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section III shall be final when the extension expires if a hearing request has not been received.

An answer or a request for hearing shall not stay the immediate effectiveness of this Order.

Dated at Rockville, Maryland, this 3rd day of December, 2003.

For the Nuclear Regulatory Commission.

J.E. Dyer,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 03–30466 Filed 12–8–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

I. Background

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

This biweekly notice includes all notices of amendments issued, or proposed to be issued from, November 14, through November 26. The last biweekly notice was published on November 25, 2003 (68 FR 66131).

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

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

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

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

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By January 8, 2004, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/doc-collections/cfr/.* If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

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

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

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

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

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

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

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudications Staff, or may be delivered to the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland, by the above date. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that petitions for leave to intervene and requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301–415–1101 or by e-mail to *hearingdocket@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and because of continuing disruptions in delivery of mail to United States Government offices, it is requested that copies be transmitted either by means of facsimile transmission to 301–415–3725 or by e-mail to *OGCMailCenter@nrc.gov.* A copy of the request for hearing and petition for leave to intervene should also be sent to the attorney for the licensee.

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

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's PDR, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC PDR Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of amendment request: August 6, 2003.

Description of amendment request: This amendment would revise the Technical Specifications (TSs) to incorporate reference to the 10 CFR 50.55a, Codes and Standards, criteria for the inservice reactor building tendon surveillance requirements, to incorporate an administrative change to the TS Definition 1.22 to be consistent with 10 CFR 20.1003, as well as other administrative corrections from previously issued TS amendments.

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 revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates reference to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in addition to the existing criteria of Regulatory Guide 1.35. This change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirements specified in 10 CFR 50.55a(b)(2)(vi). These regulatory requirements and the associated surveillance program ensure that the reactor building tendon prestressing system is capable of maintaining the structural integrity of the containment during operating and accident conditions. The reactor building prestressing system is not an initiator of any accident. Therefore, this change is not related to the probability of any accident previously evaluated. This change ensures that the containment tendon surveillance program addresses the appropriate regulatory criteria. This change does not result in any reduction in the effectiveness of the existing surveillance program. The tendon surveillance program will continue to ensure that the containment structure is capable of performing its intended safety function in the event of a design basis accident. Therefore, this change has no affect on the consequences of an accident previously evaluated.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations. 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 revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates references to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in addition to the existing criteria of Regulatory Guide 1.35. This change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirement specified in 10 CFR 50.55a(b)(2)(vi). The proposed Technical Specification change does not result in any reduction in effectiveness of the existing tendon surveillance program. The tendon surveillance program will continue to satisfy the applicable Technical Specification and regulatory required criteria, thus ensuring that the containment structure will perform its design safety function. This change has no affect on the design and operation of plant structures, systems, and components. This change does not introduce any new accident precursors and does not involve any alterations to plant configurations, which could initiate a new or different kind of accident.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations.

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

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

Response: No.

The proposed revision to Technical Specification 4.4.2.1 and associated Bases Section incorporates reference to the criteria of 10 CFR 50.55a, ''Codes and standards,'' in addition to the existing criteria of Regulatory Guide 1.35. This change provides consistency between the Technical Specification tendon surveillance program criteria and the regulatory requirement specified in 10 CFR 50.55a(b)(2)(vi). The containment examination and inspection requirements specified in 10 CFR 50.55a(b)(2)(vi) meet the same standards as the criteria specified in Regulatory Guide 1.35. The proposed Technical Specification change does not result in any reduction in effectiveness of the existing tendon surveillance program. The tendon surveillance program will continue to satisfy the applicable Technical Specification and regulatory required criteria, thus ensuring that the containment structure will perform its design safety function in accordance with existing margins of safety for containment integrity.

The proposed changes to Technical Specification Definition 1.22, Technical Specification 3.1.6.6 and associated Bases, and Technical Specification 3.24 Bases are only administrative changes or corrections and have no affect on plant design or operations.

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

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

Attorney for licensee: Edward J. Cullen, Jr., Esquire, Vice President, General Counsel and Secretary, Exelon Generation Company, LLC, 300 Exelon Way, Kennett Square, PA 19348 *NRC Section Chief:* Richard J. Laufer.

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

Date of amendments request: August 22, 2003.

Description of amendments request: The amendments would revise three different sections in the Updated Final Safety Analysis Report (UFSAR) for PVNGS [Palo Verde Nuclear Generating Station], Units 1, 2, and 3. This request would revise the sections of the UFSAR which describe the maximum fuel pin pressurization criteria used for fuel handling accident safety analyses. This change is necessitated due to the combination of higher core burnup designs, fuel which contains erbia poison, and the recent introduction of ZIRLO cladded fuel to the PVNGS reactors.

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

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

The proposed change would revise sections of the PVNGS UFSAR, which describe the maximum fuel pin pressurization criteria used for fuel handling accident safety analyses.

No additional equipment is being added as a result of the proposed change. None of the failure modes and effects analyses are impacted by the proposed change since no structures, systems, or components (SSCs) are being modified, system lineups remain the same, and operator actions for fuel handling accident are not changing. No manual actions are being substituted for automatic actions. The SSCs relied upon to mitigate the event are not changing. Specifically, the fuel building, BOPESFAS (Balance of Plant-Engineered Safety Features Actuation System), radiation monitor setpoints, etc. . . are not impacted. The methodology changes will have no impact on the likelihood of a malfunction of any SSCs.

No departures from the design or testing and performance standards outlined in any 10 CFR [Part] 50, Appendix A, General Design Criteria (GDC) will result from the proposed activity. The proposed UFSAR changes will not make any SSCs more likely to fail (no direct effects). Even with higher fuel pin pressures, the use of ZIRLO cladding provides more margin to design stress limits (liftoff pressure) than Zircaloy cladding. Regardless of the fuel type (and hence cladding type), the design stress and code allowable limits will not be exceeded. Palo Verde Nuclear Generating Station (PVNGS) ''Fuel Mishandling Accident Evaluation with ZIRLO Fuel Rods'' concluded that the analysis of record for fuel handling events involving fuel assemblies containing ZIRLO cladding would remain bounding. No physical changes to any SSCs will be performed as a result of the proposed changes. In addition, system/equipment redundancy requirements are maintained with the proposed UFSAR changes.

Fuel handling accident analyses must ensure doses at the site boundary and control room remains well within 10 CFR Part 100 and 10 CFR [Part] 50 Appendix A, GDC 19 exposure guideline. Restricting the peak assembly average fuel pin pressure to <1200 psig will still result in acceptable doses. Therefore, no indirect effects on SSCs associated with dose limitations are impacted.

Consequences mean dose at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room; therefore, an increase in consequences must involve an increase in radiological doses to the public or to control room operators. No changes to the dose exposure as a result of a fuel handling accident are proposed for the methodology change and Regulatory Guide 1.25 deviation requested. Therefore, there are no radiological consequence changes for this event.

The fuel handling accident event does involve fuel barrier failure and does involve consequences, however no changes to the fuel handling dose calculation are required since the decontamination factor will remain unchanged even with maximum fuel pin pressure exceeding 1200 psig. Activities affecting on-site dose consequences that may require prior NRC approval are those that impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

The proposed change does not modify any operator actions and hence will not impede required actions inside or outside the control room to mitigate the consequences of reactor accidents. The proposed change will not prevent or degrade the effectiveness of actions described or assumed in an accident discussed in the UFSAR. The proposed change does alter assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR, however the altered assumption is a methodology change. If the proposed methodology change were not applied, the calculated dose would increase. The peak assembly average pin pressure concept would allow the decontamination factor (DF) to remain the same and therefore consequences would remain unchanged. The proposed change does not play a direct role in mitigating the radiological consequences of an accident described in the UFSAR. The radiological consequences of the accident described in the UFSAR are bounding for the proposed activity (*e.g.*, the results of the UFSAR analysis bound those that would be associated with the proposed change).

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

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

The accident affected by the proposed change is the fuel handling accident (UFSAR Section 15.7.4). The proposed change does not involve any new equipment and does not operate any existing equipment in a different or more severe manner than what has previously been analyzed. PVNGS evaluations concluded all analyses of record for fuel handling events involving fuel assemblies containing ZIRLO cladding will remain bounding. The material strength of ZIRLO is significantly higher than that for Zircaloy-4. Since the allowable stresses for ZIRLO cladding are significantly higher than for Zircaloy-4, the same number of fuel rods (or fewer) will be damaged by the same accident scenarios as previously evaluated. Regardless of the fuel type (and hence cladding type), the design stress and code allowable limits will not be exceeded. Slight changes in the maximum fuel pin pressure during fuel movement will have no impact on the possibility of creating an accident of a different type as long as the design pressure structural limits of the fuel assembly are not approached. PVNGS calculation documents minimum liftoff pressures will not be challenged regardless of the fuel type or cladding type. Maintaining peak assembly average fuel pin pressure below 1200 psig will not challenge the liftoff pressure design basis limit for the cladding. The peak pin internal pressures for the hot rods never exceed the clad liftoff pressure and therefore the fuel pins will not be more likely to fail. Vendor calculation shows that the ZIRLO cladding fuel design results in a greater margin to the design pressure limit of the fuel cladding and also documents liftoff pressures are not exceeded for PVNGS fuel designs.

The design function of the SSCs required to function during a fuel handling accident is to provide protection to ensure fuel damage is limited to 236 fuel pins (one fuel assembly) and ensuring doses do not exceed established limits. These are indirect affects. This change will not make a SSC more likely to fail (no direct affects). In fact, ZIRLO cladding fuel is less likely to fail than the original Zircaloy-4 cladding fuel. No physical changes to the SSCs will be performed as a result of the proposed change. This proposed change does not change the failure modes for the SSCs required to operate for the fuel handling accident. The cladding calculations document design stress or code allowable limits will not be exceeded. Hence, system/equipment redundancy requirements are maintained. Fuel handling accident analyses must ensure doses at the site boundary remain within acceptable design limits. The cladding calculations document fuel pin pressures do not exceed the design pressure ratings for the fuel assembly. Therefore, no indirect effects on SSCs associated fuel clad pressure boundary exist. None of the failure modes and effects analyses are impacted by this methodology change since no SSCs are being modified, system lineups remain the same, and operator actions are not changing.

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

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

The Nuclear Reactor Regulation (NRR) Safety Evaluation, ''Related to Task Interface Agreement 99–03 Regarding Potential Nonconservative Assumptions for Fuel Handling Accident, McGuire Nuclear Station, dated November 24, 1999,'' states in part, ''The NRR staff has concluded that the increased rod pressures associated with extended bumup fuel can be expected to decrease the value of the iodine DF. However, the NRR staff believes that the iodine DF value of 100 provided in Regulatory Guide 1.25 has sufficient margin to compensate for the increases in rod gas pressures at current allowable bumup levels and for the expected increases in gap release fractions. Conservatisms in the assessment of the amount of fuel damage provide additional margin. Design basis fuel handling accidents are not considered to have a high risk significance. On the basis of these findings, the staff concludes that there is reasonable assurance that adequate protection of the public from the effects of design basis fuel handling accidents involving fuel with peak rod average bumups as high as 62 GWD/MTU will continue.''

To assess the margin of safety, the methodology specified in Regulatory Guide 1.183, [''']Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,['''] was evaluated. This regulatory guide suggests a DF of 200 for iodine. This DF is well above the DF of 100 specified by Regulatory Guide 1.25.

APS [Arizona Public Service] proposes that ample margin is retained to justify the continued used [use] of an overall decontamination factor of 100 at a peak assembly average fuel pin pressure of 1200 psig.

Therefore, APS has concluded that the proposed license amendment request does not involve a significant reduction in a margin of safety.

Based on the above, APS concludes that the [activities associated with] the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92 [''Issuance of Amendment,''] (c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendments request: September 17, 2003.

Description of amendments request: The amendments would revise sections of the Technical Specifications (TS) to support replacement of the part length control element assemblies (PLCEAs) with a new design that contains neutron absorber over the entire control section of the CEA. The replacements are referred to as part strength control element assemblies (PSCEAs). Additionally, a change is proposed to TS 3.1.5—''Control Element Assembly (CEA) Alignment,'' Condition B, to eliminate a potential condition which could cause an unwarranted plant shutdown.

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

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

The physical difference between the 4-finger full strength control element assemblies (FSCEAs) and the PSCEAs involves using Inconel rather than B_4C (boron carbide) over 100% of the active control section of each CEA finger. In addition, the PSCEAs use Inconel tubing to encase solid Inconel slugs, which cover the entire control section of the control element assembly (CEA). The current PLCEAs (also have only 4-fingers) use solid Inconel rods for only the lower half of each finger and B_4C pellets in the top 15 inches (10%) of the control section of the CEA. Although failure of the solid Inconel region due to neutron fluence would be less likely than a typical clad design, the differences in swelling between the Inconel slugs encased by Inconel clad for the PSCEAs will be minor and result in a minimal impact on clad integrity. With the exception of the neutron absorber, the cladding design used for the PSCEAs is similar to the cladding of the full strength CEAs (FSCEAs). The geometry, cladding materials, and the spider assembly that supports the CEA fingers are essentially the same for the 4-finger FSCEAs and the PSCEAs. The principal difference results from the Inconel slugs contained in the PSCEAs being heavier than the B_4C pellets used in the FSCEAs. Even though the weight of a 4-finger PSCEA is greater than the weight of a 4-finger PLCEA or a 4-finger FSCEA, this weight difference is bounded by the 12-finger FSCEAs which are operated by the same CEA drive mechanism system.

The PSCEAs use Inconel as a neutron absorber in the entire control section of each CEA finger and will be operationally used the same way as the PLCEAs. In particular, the insertion restraints that are defined by the power dependent insertion limits (PDILs) for the PLCEAs will remain the same for the PSCEAs. This existing requirement will not result in any significant operational impact on the PSCEAs since the solid Inconel cylinder in the bottom 50% (operating range of the PDILs) of the PLCEAs has essentially the same reactivity worth as that of the PSCEAs.

In addition, renaming the full length CEAs and part length CEAs to full strength CEAs and part strength CEAs, respectively, and providing definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

The PLCEAs are not credited in the accident analyses for accident mitigation. The PSCEA design eliminates an accident scenario involving the insertion of a PLCEA past the PDIL, which results in an axial shift in power due to the upper region of the PLCEAs which has no neutron absorber. This condition will not occur with the PSCEAs because they are filled with neutron absorber over 100% of the control section of each finger.

Concerning TS Limiting Condition for Operation (LCO) 3.1.5, Condition B, proposed change; there are three position indicator channels available for each CEA. Current TS Bases state that, ''At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA.'' Additionally the TS Bases states, ''If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8.'' The TS Bases make no restriction or condition limiting only one CEA within a subgroup to having only one CEA position indication channel. Current analyses already assume that more than one CEA in a subgroup could have only one position indicator OPERABLE. Modifying the wording for Condition B, of LCO 3.1.5, will not affect the likelihood or consequences of a CEA drop, slip, ejection, or misalignment. This change will still require at least one position indication channel be available for each CEA.

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

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

The proposed changes do not introduce any new mode of plant operation and the PSCEAs, like the PLCEAs, are not relied upon for accident mitigation. The PSCEAs will be operated in exactly the same manner in which the PLCEAs are operated. The existing operating restrictions for the PLCEAs will apply to the PSCEAs. In particular, the power dependent insertion limit (PDIL) restrictions identified in the Core Operating Limits Report (COLR) will remain the same for the PSCEAs. The PSCEA design uses Inconel over the entire control section of each CEA finger, which will prevent the potential undesired flux redistribution currently associated with the misoperation of PLCEAs. Therefore, the analysis associated with the undesired flux redistribution misoperation for the PLCEAs will be eliminated from PVNGS [Palo Verde Nuclear Generating Station] safety analyses. PSCEA misoperation events are bounded by the existing PLCEA and FSCEA misoperation safety analyses.

In addition, renaming (within the Technical Specifications) the ''full length CEAs'' and ''part length CEAs'' to ''full strength CEAs'' and ''part length or part strength CEAs,'' respectively, and providing a definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

Concerning TS LCO 3.1.5, Condition B proposed change, CEA position indication channels have no control function and provide input to the CEA Calculators (CEACs) and Core Protection Calculators (CPCs) for generation of a penalty factor. This change will still require at least one position indication channel be available for each CEA. Allowing Condition 'B' of LCO 3.1.5 to apply to more than one CEA per group does not create the possibility of a different type of malfunction than previously evaluated in the UFSAR [Updated Final Safety Analysis Report].

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

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

The design of the PSCEAs is very similar to the FSCEAs except for the neutron absorber within each finger of a PSCEA. The PSCEAs do not have as strong of a neutron absorber (Inconel) as that which is contained in the FSCEAs (B_4C). There is a weight difference which results from the Inconel slugs contained in the PSCEAs being heavier than the B_4C pellets used in the FSCEAs. Even though the weight of the 4-finger PSCEAs is greater than the weight of the 4-finger PLCEAs, the CEA drive mechanism and support components shall operate within their design bases. Therefore, the PSCEAs can be considered adequate for safety-related applications. Consequently, the differences in design between the current PLCEAs and the PSCEAs do not adversely impact safe operation.

The PLCEAs are not relied upon for shutdown margin or accident mitigation and no new requirements will apply to the PSCEAs. However, the design of the PSCEAs is effectively eliminating the concern associated with the insertion of the PLCEAs past the PDILs which could result in an undesirable shift in neutron flux to the top of the core due to the region within the PLCEAs that do not have neutron absorber. The PSCEAs have neutron absorber throughout their entire control section, which prevents a neutron flux shift to the top of the core if inserted past the PDIL, when compared to that of the PLCEAs.

In addition, renaming the ''full length CEAs'' and ''part length CEAs'' to ''full strength CEAs'' and ''part length or part strength CEAs,'' respectively, and providing definition for the PSCEAs will not impact the safe operation of the plant. The terminology will be appropriately changed in any related document, equipment tag, or indication on a control panel.

Concerning TS LCO 3.1.5, Condition B, proposed change, the current licensing bases already consider having more than one CEA in a CEA group with only one available position indication. The TS Bases for LCO 3.1.5, Condition B state that, ''At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA.'' Additionally the Bases states, ''If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status; or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8.'' The TS Bases make no restriction or condition limiting only one CEA within a subgroup, to having only one CEA position indication channel OPERABLE. Therefore, modifying the wording for LCO 3.1.5, Condition B, does not involve a significant reduction in the margin of safety since loss of indication to more than one CEA is already considered in the licensing bases.

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

Based on the above, APS [Arizona Public Service] concludes that the activities associated with the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92 [''Issuance of Amendment,''] (c) and, accordingly, a finding of ''no significant hazards consideration'' is justified.

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

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 7, 2003.

Description of amendment request: The licensee is proposing to revise Technical Specification (TS) Section 5.5.6, ''Pre-Stressed Concrete Containment Tendon Surveillance Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The proposed revision to TS 5.5.6 is to indicate that the Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC. The licensee has also proposed to delete the provisions of Surveillance Requirement 3.0.2 from this specification.

In addition, the licensee is proposing to revise TS 5.5.16, ''Containment Leakage Rate Testing Program,'' to add exceptions to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test 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. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes would revise Technical Specification (TS) Section 5.5.6, ''Pre-Stressed Concrete Containment Tendon Surveillance Program,'' and Section 5.5.16, ''Containment Leakage Rate Testing Program,'' for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors. The improved inspections required by the American Society of Mechanical Engineers (ASME) Code serve to maintain containment response to accident conditions, by causing the identification and repair of defects in the containment.

The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Code Section XI requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations [as] required by Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Programs,'' without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the ASME Code[-]required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed amendment does not impact any accident initiators, analyzed events, or assumed mitigation of accident or transient events. The proposed changes do not involve the addition or removal of any equipment or any design changes to the facility.

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

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

The proposed change revises the Technical Specification administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The improved inspections required by the American Society of Mechanical Engineers (ASME) Code serve to maintain containment response to accident conditions, by causing the identification and repair of defects in the containment. In addition, the change affects the frequency of visual examinations that will be performed for the concrete surface containments. The proposed change also allows those examinations to be performed during power operation as opposed to during a refueling outage. Therefore, this change updates the Technical Specifications to meet the current regulations and eliminates duplication of requirements. The safety function of the containment as a fission product barrier will be maintained.

Therefore, this proposed change does not create the possibility of an accident of a different kind than previously evaluated.

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

The proposed change revises the improved Standard Technical Specification administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The change also affects the frequency of visual examinations that will be performed for the concrete surface containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The change ensures that containment integrity [will be maintained] and ensures that the safety function of the containment as a fission product barrier will be maintained.

Therefore, this 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 amendments request involves no significant hazards consideration.

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

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: September 18, 2003.

Description of amendment request: The licensee is proposing to revise the Design Features Technical Specification 4.2, ''Fuel Storage.'' The licensee's technical specification change implements the following proposed changes:

(1) Eliminates all credit for Boraflex as a neutron absorber.

(2) Reduces the number of fuel assemblies allowed to be stored in the spent fuel pool (SFP) from 3229 to 2959. The fuel will be prohibited from being stored in 270 specific storage rack locations. This is necessary to support the elimination of all credit for Boraflex.

(3) Changes the required spent fuel pool k_{eff} to ≤0.95. This is necessary to support the elimination of all credit for Boraflex.

(4) Eliminates the design features requirements on new fuel storage, since Millstone Unit No. 1 (MP1) is a plant that has ceased power operation and will no longer receive new fuel.

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.

Accidents previously evaluated are the fuel handling accidents[, as] described in the Decommissioned Safety Analysis Report (DSAR), and a seismic event, which is considered as part of the spent fuel rack design.

Since there are no changes to plant hardware, nor any changes in how fuel is moved, there are no changes to the probability of a fuel handling accident. The consequences of a fuel handling accident are not affected, since none of the inputs to the fuel handling accident is affected.

The proposed changes affect the criticality analysis of the spent fuel storage racks. The spent fuel racks will continue to be able to perform their design function, which is to maintain the stored fuel in a sub-critical and cooled condition under all normal and postulated accident conditions. There are no physical hardware changes to the plant from these proposed changes. The revised criticality analysis submitted with these proposed changes demonstrates that fuel will be maintained in a sub-critical condition during all normal and postulated accident conditions, including the seismic event. Since there is no change in the ability of the fuel storage racks to maintain a sub-critical condition due to a seismic event, there is no change in the probability or consequences of this accident.

Reducing the amount of fuel storage is a conservative action, and the spent fuel racks were designed and licensed to allow empty, partially filled, or completely full storage racks. Thus the fuel racks will continue to be able to perform their design function to maintain the fuel in a coolable condition.

The change to the new fuel storage racks is to delete the Technical Specification requirements for the new fuel storage k_{eff} limits. Since MP1 is a plant that has ceased power operation and will no longer receive new fuel, there is no need for these Technical Specification requirements. There are no new fuel related accidents previously analyzed, therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

In summary, the proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

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

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.

Reducing the number of fuel assemblies to be stored in the pool, and discontinuing credit for Boraflex are conservative changes that do not introduce any new or different kind of failure modes.

The changes made primarily affect the nuclear criticality analysis and do not create a new or different kind of accident. Changes in eliminating Boraflex credit, restricting fuel in certain storage locations, and changing the allowable k_{eff} limit are all impacts to the nuclear criticality analysis for the SFP. 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} less than or equal to 0.95, as well as within the 10 CFR Part 50 Appendix A, ''General Design Criteria for Nuclear Power Plants,'' Criterion 62 ''Prevention of Criticality in Fuel Storage and Handling'' (Reference 6) criteria of sub-critical, have been evaluated. Criticality impacts are more appropriately discussed under the margin of safety criterion.

The change to the new fuel storage racks is to delete the Technical Specification requirements for the new fuel storage k_{eff} limits. Since MP1 is a plant that has ceased power operation and will no longer receive new fuel, there is no need for these Technical Specification requirements. Since Millstone 1 currently has no new fuel and new fuel cannot be received, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

The margin of safety relevant to the SFP is defined as (1) SFP k_{eff} remains sub-critical by an acceptable margin, and (2) the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact.

The industry and regulatory accepted value for [the] required sub-criticality margin[s] in the SFP is to ensure that the k_{eff} of the SFP remains ≤0.95 under all normal and postulated accident conditions. This is documented in the Standard Review Plan, Regulatory Guide 1.13, and ANSI/ANS–57.2, ''American National Standard Design Requirements for LWR Spent Fuel Storage Facilities at Nuclear Power Plants.'' The current MP1 Technical Specifications require a more conservative value of 0.90 for SFP k_{eff}. The proposed Design Features Technical Specification changes the maximum SFP k_{eff} from 0.90 to 0.95. This is not a significant reduction in the margin to [of] safety since the proposed value of 0.95 is consistent with the accepted regulatory guidance for [the] sub-criticality margin. The proposed criticality analysis demonstrates that the SFP k_{eff} remains ≤0.95 on a 95/95 basis under all normal and postulated accident conditions, thus the required margin of criticality safety has been maintained.

The proposed changes conservatively reduce the amount of fuel that can be stored, and therefore do not affect the SFP cooling analysis. Therefore, the spent fuel in the SFP remains adequately cooled so that the fission product barriers remain intact.

The removal of Technical Specification requirements for the new fuel storage k_{eff} limits does not affect the margin of safety since new fuel can no longer be received.

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

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

Attorney for licensee: Lilliam M. Cuoco, Esq., Senior Nuclear Counsel, Dominion Nuclear Connecticut, Inc., Rope Ferry Road, Waterford, Connecticut 06385.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; and January 27, September 22, and November 20, 2003.

Description of amendment request: The proposed amendments would revise the Technical Specifications to incorporate changes resulting from the use of an alternate source term and the implementation of several plant modifications. Publications of the Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing have already appeared in the **Federal Register** on January 22, 2002 (67FR2922) and October 14, 2003 (68FR59215). The November 20, 2003, submittal contained a revised No Significant Hazards Consideration Determination.

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:

Standards for determining whether a license amendment involves no significant hazards considerations are contained in 10CFR50.92(c). The TS [Technical Specification] changes and modifications as proposed in this LAR [license amendment request] have been evaluated in accordance with 10 CFR 50.92 and determined not to involve any significant hazards considerations.

The proposed LAR includes (1) implementing the AST [alternate source term] for accident analysis as described in Regulatory Guide 1.183; (2) removing the PRVS [penetration room ventilation system] and relaxing the SFPVS [spent fuel pool ventilation system] TS because they are no longer credited for Control Room and off-site doses; (3) revising the CRVS [control room ventilation system] to allow for a one time completion time extension on Conditions B and C when entering the conditions to support implementation of the Control Room intake/booster fan modification; (4) lowering the Reactor Building leakage rate from 0.25 w%/day to 0.20 w%/day; (5) revising the VFTP [ventilation filter testing program] radioactive methyl iodide removal acceptance criterion for SFPVS and CRVS Booster Fan trains; and (6) adoption of TSTF [Technical Specification Task Force]–51.

Plant modifications are also being proposed in concert with the proposed TS changes. They include relocating the existing Control Room outside air intake from the roof of the Auxiliary Building to the roof of the Turbine Building and installing dual intakes for each Control Room; re-routing HPI [high-pressure injection]/LPI [low-pressure injection] relief valve discharge back into the Reactor Building and replacing the existing Caustic Addition system with a passive system.

As a result of this evaluation, Duke has concluded:

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

The AST and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The AST does not affect the design or operations of the facility. Rather, the AST is used to evaluate the consequences of a postulated accident. The implementation of the AST has been evaluated in the revisions to the analysis of the design basis accidents for ONS [Oconee Nuclear Station]. Based on the results of these analyses, it has been demonstrated that, with the requested changes, the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. Therefore, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The AST and those plant systems affected by implementing the proposed changes to the TS are not assumed to initiate design basis accidents. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed TS changes and modifications do not significantly affect the mitigative function of these systems. Consequently, these systems do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The implementation of the AST, proposed changes to the TS and the implementation of the proposed modifications have been evaluated in the revisions to the analysis of the consequences of the design basis accidents for the ONS. Based on the results of these analyses, it has been demonstrated that with the requested changes the dose consequences of these events meet the acceptance criteria of 10 CFR 50.67 following the provisions of Regulatory Guide 1.183. Thus, the proposed amendment will not involve a significant reduction in the margin of safety.

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

Attorney for licensee: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005,

NRC Section Chief: John A. Nakoski.

Duke Energy Corporation, Docket No. 50–270, Oconee Nuclear Station, Unit 2, Oconee County, South Carolina

Date of amendment request: October 28, 2003.

Description of amendment request: The proposed amendment would revise the licensing basis in the Updated Final Safety Analysis Report to support installation of a passive low-pressure injection (LPI) cross connect inside containment. The proposed changes would revise the licensing basis for selected portions of the core flood and LPI piping to allow exclusion of the dynamic effects associated with postulated pipe rupture of that piping by application of leak-before-break methodology. A similar amendment was approved for Unit 1 by NRC letter dated September 29, 2003.

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

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures 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:

The proposed LAR [license amendment request] modifies the Unit 2 licensing basis to allow the dynamic effects associated with postulated pipe rupture of selected portions of the Unit 2 LPI [low-pressure injection]/Core Flood (CF) piping to be excluded from the design basis. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet GDC [General Design Criterion] 4 requirements related to environmental and dynamic effects. The proposed LAR will continue to ensure that ONS [Oconee Nuclear Station] can meet design basis requirements associated with the LPI safety function. The addition of the crossover line will enhance the ability of the control room operator to mitigate the consequences of specific events for which LPI is credited. Therefore, the proposed LAR does 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 kind of accident previously evaluated:

The proposed LAR modifies the Unit 2 licensing basis to allow the dynamic effects associated with postulated pipe rupture of selected portions of the Unit 2 LPI/Core Flood (CF) piping to be excluded from the design basis. The proposed design allowances for these selected portions of piping continue to allow the LPI system design to meet GDC 4 requirements related to environmental and dynamic effects. The systems affected by the changes are used to mitigate the consequences of an accident that has already occurred. The proposed licensing basis change does not affect the mitigating function of these systems. Consequently, these changes do not alter the nature of events postulated in the Safety Analysis Report nor do they introduce any unique precursor mechanisms. Therefore, the proposed amendment will 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 proposed licensing basis change does not unfavorably affect any plant safety limits, set points, or design parameters. The change also do [SIC] not unfavorably affect the fuel, fuel cladding, RCS [reactor coolant system], or containment integrity. Therefore, the proposed licensing basis change, which adds new design allowances associated with the passive LPI cross connect modification, do [SIC] 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: Anne W. Cottington, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20005.

NRC Section Chief: John A. Nakoski.

Energy Northwest, Docket No. 50–397, Columbia Generating Station, Benton County, Washington

Date of amendment request: November 4, 2003.

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a technical specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TSs would be eliminated, and Surveillance Requirement (SR) 3.0.4 revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated November 4, 2003.

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 allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change 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). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Thomas C. Poindexter, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005–3502.

NRC Section Chief: Stephen Dembek.

Entergy Gulf States, Inc., and Entergy Operations, Inc., Docket No. 50–458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: October 21, 2003.

Description of amendment request: The proposed change would remove MODE restrictions that currently prevent performance of Surveillance Requirements (SRs) 3.8.4.7 and 3.8.4.8 for the Division III DC electrical power subsystem while in MODE 1, 2, or 3. These surveillances verify that the battery capacity is adequate to perform its required functions. The changes would allow the performance of SR 3.8.4.7 and SR 3.8.4.8 during normal plant operations rather than only during refueling outages.

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

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

Response: No.

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

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

The out of service condition is controlled and evaluated for safety implications in accordance with 10 CFR 50.65 [''Requirements for monitoring the effectiveness of maintenance at nuclear power plants'']. The HPCS system reliability and availability are monitored and evaluated in relationship to Maintenance Rule goals to ensure that total outage times do not degrade operational safety over time.

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

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

Response: No.

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

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

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

Response: No.

The battery testing will be performed when the HPCS system is already out of service for maintenance. The out of service condition is controlled and evaluated for safety implications in accordance with 10 CFR 50.65. The batteries are not expected to be unavailable for more than 36 hours. This testing period is within the period of time that the system will already be out of service for other planned maintenance. Therefore, the battery test does not increase unavailability of the supported system or represent any change in risk above the current practice of planned system maintenance outages as currently allowed by the TS. Timing of this test has no effect on any fission product barrier.

Therefore, the propose 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: Mark Wetterhahn, Esq., Winston & Strawn, 1400 L Street, NW., Washington, DC 20005.

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 21, 2003.

Description of amendment request: The proposed amendment would revise Technical Specification Section 5.5.7, ''Steam Generator (SG) Tube Surveillance Program,'' to allow a one-time extension of the frequency for examination of the SG tubes. Specifically, the amendment would extend the examination, currently due no later than November 17, 2004, to June 17, 2006.

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.

There is no direct increase in SG leakage because the proposed change does not alter the plant design. The scope of the inspection performed during the first refueling outage subsequent to SG replacement (last outage), exceeded the technical specification requirements for the first two refueling outages combined, after replacement. More tubes were inspected than were required by the technical specifications. Indian Point 2 does not have an active SG damage mechanism and will meet the current industry examination guidelines without performing inspections during the next refueling outage. The results of the Condition Monitoring Assessment subsequent to the last outage, demonstrated that all performance criteria were met during the last operating period. The results of the aforementioned Operational Assessment show that all performance criteria will be met over the proposed operating period.

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

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

Response: No.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of the inspections performed during the last (first after SG replacement) refueling outage significantly exceeds the Technical Specification requirements for the scope of the first two refueling outages combined subsequent to SG replacement.

The proposed change does not affect the SG design, the method of operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension of the SG tube inservice inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

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.

SG tube integrity is a function of design, environmental, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. Inspections conducted prior to placing the SGs into service (pre-service inspection) and inspection during the first refueling outage following SG replacement, demonstrate that the SGs do not have fabrication damage or an active damage mechanism. The scope of those inspections significantly exceeds those required by the technical specifications. These inspection results were comparable to similar inspection results for the same model SG installed at other plants, and subsequent inspections at those plants provided results that support the extension request. The improved design of the replacement SGs also provides assurance that significant tube degradation is not likely to occur over the proposed operating period.

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

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

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

NRC Section Chief: Richard J. Laufer.

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

Date of amendment request: June 30, 2003, as supplemented by letter dated November 20, 2003.

Description of amendment request: The proposed amendment would (1) reorganize the Arkansas Nuclear One, Unit No. 2 (ANO–2) Technical Specifications (TSs) Section 6.0, Administrative Controls, (2) modify the ANO–2 Facility Operating License, and actions and surveillance requirements (SRs) of various other TSs, to support the reorganization of Section 6.0, and (3) modify several actions and SRs that are related to systems that are shared by ANO–2 and Arkansas Nuclear One, Unit No. 1 (ANO–1). These changes are being proposed so that the philosophy and location of the TSs in Section 6.0 reflect the recently approved conversion of the ANO–1 TSs to the Improved Technical Specifications (ITS) and the subsequent amendments to the ANO–1 ITS. This amendment request supersedes the previous application related to the revision of TS Section 6.0 dated January 31, 2002, as supplemented on June 26 and July 18, 2002. The January 31, 2002, application was previously noticed in the **Federal Register** on March 19, 2002 (67 FR 12602), and the June 30, 2003, application was previously noticed in the **Federal Register** on July 22, 2003 (68 FR 43385).

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

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

Response: No.

Administrative Changes

The proposed changes involve reformatting and rewording of the existing TSs. The reformatting and rewording process involves no technical changes to existing requirements. As such, the proposed changes are administrative in nature and do not impact initiators of analyzed events or assumed mitigation of accident or transient events.

Less Restrictive—Administrative Deletion of Requirements

The proposed changes relocate requirements from the TSs to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements.

More Restrictive Changes

The proposed changes provide more stringent requirements for the ANO–2 TSs. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The more stringent requirements are imposed to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis and to provide greater consistency with the ANO–1 TS and NUREG 1432.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

The control room area radiation monitor is used to support mitigation of the consequences of an accident; however, it is not considered the initiator of any previously analyzed accident. Also, the addition of the Note to allow time for testing reduces the potential for initiation of a previously analyzed accident due to reduced potential for shutdowns and startups due to incomplete or missed surveillances. As such, the proposed revision to include an allowance for testing does not significantly increase the probability of any accident previously evaluated. This change does not result in any hardware changes, but does allow operation for a limited time with an inoperable monitor for the purposes of testing. Since the capability of the control room area radiation monitor to provide the required information continues to be verified, and the time allowed for inoperability for testing is short, the change will not reduce the capability of required equipment to mitigate the event. Also, the consequences of an event occurring during the proposed operation of the unit during the allowed inoperability for testing are the same as the consequences of an event occurring while operating under the current TS Actions. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the CREVS [control room emergency ventilation system] to be inoperable due to control room boundary inoperability for a period of 24 hours.

Neither CREVS nor the control room boundary is the initiator of any accident analyzed in the SAR [Safety Analysis Report]. Therefore, this change does not result in a significant increase in the probability of an accident previously evaluated.

The CREVS and the control room boundary are intended to provide a habitable environment for the control room operators in the event of an accident that results in the release of radioactivity to the environment. The allowance to open the control room boundary intermittently is acceptable, because of the administrative controls that will be implemented to ensure that the opening can be rapidly closed when the need for control room isolation is indicated, restoring the control room habitability envelope. Allowing both CREVS trains to be inoperable for 24 hours due to an inoperable control room boundary is acceptable because of the low probability of an accident requiring control room isolation during any given 24 hour period, because entry into this condition is expected to be an infrequent occurrence, and because preplanned compensatory measures to protect the control room operators from potential hazards are implemented. Therefore, this change will not result in a significant increase in the probability [consequences] of an accident previously evaluated.

(3) An allowance will be added to allow use of a ''simulated'' or ''actual'' signal when testing the automatic isolation feature of the control room air filtration system.

The phrase ''actual or simulated'' in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an ''actual'' signal, nor does it eliminate any restriction on producing an ''actual'' signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the function of the system functional test remains unaffected the change does not involve a significant increase in the consequences of an accident previously evaluated.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the individual volume is greater than 17,446 gallons will be added. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

The AC Sources are used to support mitigation of the consequences of an accident and can be involved in the initiation of the accident analyzed in SAR. Equipment powered by the AC Sources, which may be considered as an initiator, continues to be assured of electrical power. The proposed increased restoration time involves parameters unrelated to initiating the failure of the AC Sources. As such the proposed time allowance for restoration of limited levels of readiness parameter degradation will not increase the probability of any accident previously evaluated. The proposed changes allow additional time for restoration of parameters that have been identified as not immediately affecting the capability of the power source to provide its required safety function. The identified parameters are capable of being replenished during operation of the diesel generators, and the short additional allowable action time continues to provide adequate assurance of operable required equipment. Therefore, this change does not involve a significant increase in the probability of or the consequences of any accident previously evaluated.

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

The testing of diesel generator fuel oil is not considered an initiator, or a mitigating factor, in any previously evaluated accident. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is being added. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

This change does not result in any changes in hardware or methods of operation. The change allowing the absence of the STA or the radiation protection technician is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

This change does not result in any changes in hardware or methods of operation. The change in the support relationship between the STA and the control room staff is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

This change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternatives to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

The controls for access to a high radiation area are not considered as initiators, or as a mitigation factor, in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

The testing of diesel generator fuel oil is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

Notification of the Vice President, Operations ANO when a safety limit is violated is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

This change does not result in any changes in hardware or methods of operation. The change in date for submittal of ''after the fact'' information is not considered in the safety analysis, and cannot initiate or affect the mitigation of an accident in any way. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(13) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

The extension of the testing frequency, up to 25% of the test interval, is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(14) A change that allows the OSRC [Onsite Safety Review Committee] review of the desirability of maintaining a channel in the bypassed condition to be at or before the next regularly scheduled meeting.

The proposed change is not considered an initiator or a mitigating factor in any previously evaluated accident. Therefore, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

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

Response: No.

Administrative Changes

The proposed changes do not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operations. The proposed changes will not impose any different requirements.

Less Restrictive—Administrative Deletion of Requirements

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operations. The proposed changes will not impose any different requirements and adequate control of the information will be maintained.

More Restrictive Changes

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed changes do impose different requirements. However, these changes do not impact the safety analysis and licensing basis.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the control room ventilation system (CREVS) to be inoperable due to a control room boundary inoperability for a period of 24'hours.

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) An allowance will be added to allow use of a ''simulated'' or ''actual'' signal when testing the automatic isolation feature of the control room air filtration system.

The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the individual volume is greater than 17,446 gallons will be added. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure operable safety equipment is available. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is proposed. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the STA and radiation protection staffing positions and does not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the support relationship the STA provides the control room staff and does not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for submittal of information and does not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternates to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

No changes are proposed in the manipulation of the plant structures, systems, or components, or in the design of the plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will impact only the administrative requirements for submittal of information and does not directly impact the operation of the plant. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(13) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(14) A change that allows the OSRC review of the desirability of maintaining a channel in the bypassed condition to be at or before the next regularly scheduled meeting.

No changes are proposed that result in the manipulation or the design of plant structures, systems, or components. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

Response: No.

Administrative Changes

The proposed changes will not reduce the margin of safety because they have no impact on any safety analysis assumptions. The changes are administrative in nature.

Less Restrictive—Administrative Deletion of Requirements

The proposed changes will not reduce a margin of safety because they have no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the TSs to other license basis documents, which are under licensee control, are the same as the exiting TSs. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements.

More Restrictive Changes

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

(a) increasing the scope of the specification to include additional plant equipment,

(b) providing additional actions,

(c) decreasing restoration times, or

(d) imposing new surveillances.

The changes are consistent with the safety analysis and licensing basis.

Less Restrictive Changes

(1) A note will be added that allows three (3) hours to perform the channel functional test on the control room radiation monitors without entering the associated Actions.

The margin of safety for the control room area radiation monitor is based on availability and capability of the instrumentation to provide the required information to the operator. The frequency is based on unit operating experience that demonstrates channel failure is rare, and on the use of less formal but more frequent checks of channels during normal operational use of the displays associated with the required channels. Therefore, the availability and capability of the control room area radiation monitor continues to be assured by the proposed Surveillance Requirements and this change does not involve a significant reduction in a margin of safety.

(2) This change will allow the control room boundary to be opened intermittently under administrative controls, and will allow both trains of the control room ventilation system (CREVS) to be inoperable due to control room boundary inoperability for a period of 24 hours.

This change does not involve a significant reduction in a margin of safety since: (1) Administrative controls will be in place to ensure that an open control room boundary can be rapidly closed when a need for control room isolation is indicated; and (2) an inoperable control room boundary that renders both trains of CREVS inoperable is an infrequent occurrence, the probability of an accident requiring control room isolation during any given 24 hour period is low, and preplanned compensatory measures to protect the control room operators from potential hazards are implemented.

(3) An allowance will be added to use a simulated or actual signal when testing the automatic isolation feature of the control room air filtration system.

Use of an actual signal instead of the existing requirement which limits use to a simulated signal, will not affect the performance of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between ''actual'' or ''simulated'' signals. Therefore, the change does not involve a significant reduction in a margin of safety.

(4) An allowance for the diesel fuel storage tanks to contain less than 22,500 gallons of fuel for up to 48 hours as long as the

individual volume is greater than 17,446 gallons. The lower value when summed with the contents of the other tank ensures six days of fuel oil is available. During the 48 hours, the diesel generator is capable of performing its intended function. There is a low probability that an event would occur for which the diesel generator would be required during this short period of time when the lower fuel oil volume is allowed.

The parameter limits provide substantial margin to the parameter values that would be absolutely necessary for diesel generator operability. When the parameters are less than their limits this margin is reduced. However, the availability of AC Sources continues to be assured since the allowed time for parameters to be less than their limits is short and the allowed levels for the parameters are adequate to provide the immediately needed power availability. Further, the parameters can be restored to within limits during the proposed time provided should they be required. Therefore, this change does not result in a significant reduction in [a] margin of safety.

(5) Seven days will be allowed to restore the stored diesel fuel oil total particulates to within the required limits prior to declaring the associated diesel inoperable.

The proposed change allows the stored diesel fuel oil total particulates to be outside the required limits for seven days before declaring the associated diesel inoperable. The presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine. In addition, particulate concentration is unlikely to change significantly between surveillance intervals (31 days). The seven day allowance provides an appropriate backstop to ensure the particulate level is restored to within limits in a reasonable time period. Since the diesel is still capable of performing its function the margin of safety is not reduced.

(6) An allowance for the person who is satisfying the requirement of the radiation protection staff position and for the person filling the Shift Technical Advisor (STA) position to be vacant for not more than two hours in order to provide for unexpected absences is proposed. This is consistent with the allowance permitted for the control room operator as reflected in existing TSs.

The margin of safety is not dependent on the presence of the STA or the radiation protection technician. Therefore, this change does not involve a significant reduction in a margin of safety.

(7) The STA will be allowed to support the shift crew rather than only the shift supervisor. This provides more flexibility and does not dilute the function of the STA.

The margin of safety is not dependent upon who the STA supports. Therefore, this change does not involve a significant reduction in a margin of safety.

(8) The Occupational Radiation Exposure Report will be submitted by April 30 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(9) An allowance is proposed that will revise the high radiation areas to include additional previously approved methods for implementation of alternatives to the ''control device'' or ''alarm signal'' requirements of 10 CFR [Part] 20. These alternatives provide adequate control of personnel in high radiation areas as evidenced by NRC issuance of NUREG–1432.

The requirements for control of high radiation areas provide for the use of alternates to the ''control device'' or ''alarm signal'' requirements of 10 CFR 20.1601. This change provides such alternative methods for controlling access. These methods and additional administrative requirements have been determined to provide adequate controls to prevent unauthorized and inadvertent access to such areas. Therefore, this change does not involve a significant reduction in a margin of safety.

(10) An allowance to require periodic testing of stored fuel for the particulates only is proposed.

The testing of stored diesel generator fuel oil is revised to require the periodic testing of the stored fuel oil only for particulates (replacing the periodic testing per ASTM–D975) once every 31 days. The change reflects industry-standard acceptable DG fuel oil testing programs. Over the storage life of ANO–2 DG fuel oil, the properties tested by ASTM–D975 are not expected to change and performing these tests once on the new fuel oil provides adequate assurance of the proper initial quality of fuel oil. The periodic testing for particulates monitors a parameter that reflects degradation of fuel oil and can be trended to provide increased confidence that the stored DG fuel oil will support DG operability. Therefore, this change does not involve a significant reduction in a margin of safety.

(11) The removal of the requirement to notify the Vice President, Operations ANO within 24 hours of violating a safety limit.

The margin of safety is not dependent upon notification of the Vice President, Operations ANO upon the violation of a TS safety limit. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

(12) The Radioactive Effluent Release Report will be submitted by May 1 of each calendar year instead of prior to March 1.

The margin of safety is not dependent on the submittal of information. Therefore, this change does not involve a significant reduction in a margin of safety.

(13) A change that allows a 25% extension of the frequency in accordance with SR 4.0.2 for the integrated leak tests of each system outside containment that could contain highly radioactive fluids.

The proposed allowance allows a possible increase in performance interval. However, the test will still be performed at reasonable intervals to ensure the intent of the surveillance is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

(14) A change that allows the OSRC review of the desirability of maintaining a channel in the bypassed condition to be at or before the next regularly scheduled meeting.

The proposed change allows the OSRC review to occur earlier than previously required if an OSRC meeting is called before the next regularly scheduled meeting. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

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

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: October 22, 2003.

Description of amendment request: The licensee proposes to change the existing pressure/temperature limits (P/T) from 16 to 32 effective full power years (EFPY). In addition, the maximum heatup rate will be changed to 60 °F per hour and the maximum cooldown rate to 100 °F per hour for all reactor coolant system temperatures. For inservice hydrostatic pressure and leak testing, the maximum heatup and cooldown rates will be changed to 60 °F and 100 °F, respectively.

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 analyzed?

Response: No.

The probability of occurrence of an accident previously evaluated for Waterford 3 is not altered by the proposed amendment to the TSs [Technical Specifications]. The accidents currently analyzed in the Waterford 3 Final Safety Analysis Report (FSAR) remain the same considering the results of the proposed changes to the P/T limits and the LTOP [low temperature overpressure] enable temperature. The new P/T and LTOP enable temperature limits were based on the NRC [Nuclear Regulatory Commission] accepted methodologies along with the ASME [American Society of Mechanical Engineers] Code [Boiler and Pressure Vessel Code] alternatives. The proposed changes do not impact the integrity of the reactor coolant pressure boundary (RCPB) (*i.e.*, there is no change to the operating pressure, materials, loadings, etc.). The proposed change does not affect the probability nor consequences of any design basis accident (DBA). The proposed P/T limit curves, maximum heatup and cooldown rates, and the LTOP enable temperature are not considered to be an initiator or contributor to any accident currently evaluated in the Waterford 3 FSAR. The new limits ensure the long term integrity of the RCPB.

Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life. During the spring 2002 Waterford 3 refueling outage, a reactor vessel specimen capsule was withdrawn and analyzed to predict the fracture toughness requirements using projected neutron fluence calculations. For each analyzed transient and steady state condition, the allowable pressure is determined as a function of reactor coolant temperature considering postulated flaws in the reactor vessel beltline, inlet nozzle, outlet nozzle, and closure head.

The predicted radiation induces ''RT_{NDT} was calculated using the respective reactor vessel beltline materials copper and nickel contents and neutron fluence applicable to 32 EFPY including an estimated increase in flux due to proposed power uprates. The RT_{NDT} and, in turn, the operating limits for Waterford 3 were adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. Therefore, new operating limits will be established which are represented in the revised operating curves for heatup/criticality, cooldown, and inservice hydrostatic testing contained in the TSs.

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 changes to the P/T and LTOP enable temperature will not create a new accident scenario. The requirements to have P/T limits and LTOP protection are part of the licensing basis for Waterford 3. The approach used to develop the new P/T limits and LTOP enable temperature meets NRC and ASME regulations and guidelines. The data analysis for the vessel specimen removed during the last Waterford 3 refueling outage confirms that the vessel materials are responding as predicted.

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 existing P/T curves and LTOP enable temperature in the TSs are reaching their expiration period for the number of years at effective full power operation. The revision of the P/T limits and curves will ensure that Waterford 3 continues to operate within the operating margins allowed by 10 CFR 50.60 and the ASME Code. The material properties used in the analysis are based on results established through Westinghouse material reports for copper and nickel content. The application of ASME Code Case N–641 presents alternative procedures for calculating P/T and LTOP temperatures in lieu of that established for ASME Section XI, Appendix G–2215. The Code alternative allows certain assumptions to be conservatively reduced. However, the procedures allowed by Code Case N–641 still provide significant conservatism and ensure an adequate margin of safety in the development of P/T operating and pressure test limits to prevent non-ductile fractures.

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

NRC Section Chief: Robert A. Gramm.

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

Date of amendment request: September 8, 2003.

Description of amendment request: The proposed change allows entry into a mode or other specified condition in the applicability of a Technical Specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Section 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4, exceptions in individual TSs, would be eliminated, and Surveillance Requirement (SR) 3.0.4 revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated September 8, 2003.

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 allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change 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). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Mr. Edward Cullen, Vice President & General Counsel, Exelon Generation Company, LLC, 2301 Market Street, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

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

Date of application for amendments: September 26, 2003.

Description of amendment request: The proposed amendment would modify the fire protection plan (FPP). The change to the FPP would allow converting the existing carbon dioxide (CO_2) fire suppression systems, located in the cable spreading room (CSR) and each of the four emergency diesel generator rooms, from automatic to manual actuation systems.

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

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

Response: No.

The proposed activity involves changing the actuation of the carbon dioxide (CO_2) fire suppression systems from automatic to manual. With the exception of the Emergency Diesel Generator (EDG) CO_2 system itself, the proposed activity does not result in any physical changes to safety-related structures, systems, or components (SSCs), or the manner in which safety-related SSCs are operated, maintained, modified, tested, or inspected. The EDG CO_2 system is safety related due to a potential common mode effect on all four EDGs in the event of a seismic event. Eliminating the automatic actuation function of the EDG CO_2 system will thereby eliminate a potential common mode effect on the EDGs. The proposed activity does not degrade the performance or increase the challenges of any safety-related SSCs assumed to function in the accident analysis. As a result, the proposed activity does not introduce any new accident initiators. In addition, fires are not an accident that is previously evaluated. Regardless, the proposed activity does not change the probability of a fire occurring since fire ignition frequency is independent of the method of fire suppression in the room. The consequences of the proposed activity are bounded by the fire safe shutdown analysis, which assumes fire damage throughout the affected fire area. The fire safe shutdown analysis for each of the areas addressed by the proposed activity demonstrates that safe shutdown can be accomplished assuming that no fire suppression is available. In addition, the removal of the automatic discharge capability of the CO_2 system in each of the EDG rooms significantly reduces the potential for an inadvertent discharge to shutdown the EDG if needed for non-fire accident conditions. Similarly, removal of the automatic discharge feature in the CSR significantly reduces the potential for an inadvertent discharge that would require (by procedure) immediate shutdown of both units, and the potential migration of CO_2 into the main control room or other areas. In the future, CO_2 discharge will only occur as a deliberate action to the most extreme fires, as one element of an overall graded approach to fire fighting in the affected areas.

Therefore, changing the actuation of the CO_2 fire suppression systems from automatic to manual 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 activity involves changing the actuation of the CO_2 fire suppression systems from automatic to manual. With the exception of the Emergency Diesel Generator (EDG) CO_2 system itself, the proposed activity does not result in any physical changes to safety-related structures, systems, or components (SSCs), or the manner in which safety-related SSCs are operated, maintained, modified, tested, or inspected. The proposed activity does not degrade the performance or increase the challenges of any safety-related SSCs assumed to function in the accident analysis. As a result, the proposed activity does not introduce nor increase the number of failure mechanisms of a new or different type than those previously evaluated. The fire safe shutdown analysis assumes fire damage throughout the area consistent with a complete lack of fire suppression capability. The elimination of the potential for inadvertent actuation accomplished by changing the CO_2 systems from automatic to manual prevents the CO_2 systems from creating a challenge to existing accidents.

Therefore, changing the actuation of the CO_2 fire suppression system from automatic to manual does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed activity involves changing the actuation of the CO_2 fire suppression systems from automatic to manual. With the exception of the Emergency Diesel Generator (EDG) CO_2 system itself, the proposed activity does not result in any physical changes to safety-related structures, systems, or components (SSCs), or the manner in which safety-related SSCs are operated, maintained, modified, tested, or inspected. The proposed activity does not degrade the performance or increase the challenges of any safety-related SSCs assumed to function in the accident analysis. The proposed activity does not impact plant safety since the conclusions of the fire safe shutdown analysis remain unchanged.

Therefore, changing the actuation of the CO_2 fire suppression system from automatic to manual does not involve a significant reduction in a margin of safety.

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

Attorney for Licensee: Mr. Edward Cullen, Vice President and General Counsel, Exelon Generation Company, LLC, 2301 Market Street, S23–1, Philadelphia, PA 19101.

NRC Section Chief: James W. Clifford.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: August 25, 2003.

Description of amendment request: The proposed amendment would allow extension of the current Emergency Diesel Generator (EDG) Technical Specifications allowed outage time (AOT) from 72 hours to a period of 14 days. This proposal would be supported by permanently installing a non-safety-related supplemental emergency power system (SEPS).

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

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

The proposed changes do not involve a change in the operational limits or physical design of the electrical power systems, particularly the emergency power systems. The proposed changes do not change the function or operation of plant equipment or affect the response of that equipment if called upon to operate. The proposed AOT extensions to allow for additional operational flexibility will not cause a significant increase in the probability or consequences of an accident previously evaluated. In actuality, the installation of the SEPS will have an overall net reduction in core damage frequency. The AOT extensions will lessen the burden of time pressure to quickly determine the cause of failure and perform corrective actions without needing to place the plant in a transient to shutdown because of a short allotted AOT.

A Probabilistic Risk Assessment (PRA) has been performed to quantitatively assess the risk impact of an increase in the Allowed Outage Time. The proposed change results in a significant decrease in core damage frequency (CDF). Large Early Release Frequency (LERF) is dominated by containment bypass and containment isolation failures and remains relatively unchanged by the addition the SEPS combined with a 14-day AOT.

Based on the above, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

The proposed changes do not involve a change in the operational limits or physical design of the electrical power systems, particularly the emergency power systems. The proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The SEPS and interfacing components with the safety-related busses have been designed to ensure independence and separation, particularly during faulted conditions. As such, no new failure modes are being introduced. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes.

The proposed amendment extends the Allowed Outage Times for restoring an inoperable EDG to OPERABLE status and extends the period for operability verification of redundant features to allow for minor repair prior to placing the plant in a shutdown transient. The proposed amendment will not result in changes to the type of corrective or preventive maintenance activities associated with the EDGs. Plant operating procedures and the procedures used to respond to abnormal or emergency conditions will be enhanced with the option to use the SEPS when deemed necessary. Assumptions made in the safety analysis related to EDG availability will also remain unchanged. Performance of certain maintenance activities at power requires an evaluation to assure plant safety is maintained or enhanced, which would include evaluation for new or different plant conditions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes do not involve a change in the operational limits. The proposed changes do not change the function or operation of plant equipment or affect the response of that equipment if it is called upon to operate. The performance capability of the emergency diesel generators will not be affected. Installation of the SEPS will have an overall net reduction in core damage frequency. Emergency diesel generator reliability and availability will be improved by implementation of the proposed changes. In addition, administrative controls will ensure there are adequate compensatory measures that can be and will be taken during extended EDG maintenance activities to reduce overall risk. The results of the PRA performed to quantitatively assess the risk impact of an increase in the Allowed Outage Time indicate the proposed change results in a significant decrease in core damage frequency (CDF) by up to 30 percent. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

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

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

NRC Section Chief: James W. Clifford.

FPL Energy Seabrook, LLC, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: October 6, 2003.

Description of amendment request: The proposed amendment would revise the Seabrook Station licensing basis to implement the alternative source term (AST) methodology of Regulatory Guide (RG) 1.183 through reanalysis of the radiological consequences of a number of the Updated Final Safety Analysis Report Chapter 15 accidents. Further, having revised the licensing basis, the amendment would also revise the definition of dose equivalent I–131 in Technical Specifications Section 1.12.

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

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

Alternative source term calculations have been performed that demonstrate the dose consequences remain below limits specified in NRC [Nuclear Regulatory Commission] Regulatory Guide 1.183 (July 2000) and 10CFR50.67. The proposed change does not modify the physical design or operation of the plant. The use of AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of the accident. AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed change to the Technical Specifications, demonstrate that the dose consequences of these limiting events are all within the regulatory limits. The proposed Technical Specification change to the definition of dose equivalent I–131 is consistent with the implementation of AST and the requirements of RG 1.183 (July 2000).

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

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

The proposed change does not affect any plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change. The alternative source term and the dose equivalent I–131 definition change do not have the capability to initiate accidents. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed implementation of the alternative source term methodology is consistent with NRC RG 1.183 (July 2000). The Technical Specification change to the definition of dose equivalent I131 is consistent with the implementation of AST and the requirements of RG 1.183 (July 2000). Conservative methodologies, per the guidance of RG 1.183 (July 2000), have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the Control Room are within the corresponding regulatory limits of RG 1.183 (July 2000) and 10CFR50.67. The margin of safety for the radiological consequences of these accidents is considered to be that provided by meeting the applicable regulatory limits, which are set at or below the 10CFR50.67 limits. An acceptable margin of safety is inherent in these limits.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment requests: September 26, 2003.

Description of amendment requests: The proposed change allows entry into a mode or other specified condition in the applicability of a Technical Specification (TS), while in a condition statement and the associated required actions of the TS, provided the licensee performs a risk assessment and manages risk consistent with the program in place for complying with the requirements of 10 CFR 50.65(a)(4). Limiting Condition for Operation (LCO) 3.0.4 exceptions in individual TS would be eliminated, and Surveillance Requirement (SR) 3.0.4 revised to reflect the LCO 3.0.4 allowance.

This change was proposed by the industry's Technical Specification Task Force (TSTF) and is designated TSTF–359. The NRC staff issued a notice of opportunity for comment in the **Federal Register** on August 2, 2002 (67 FR 50475), on possible amendments concerning TSTF–359, 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 April 4, 2003 (68 FR 16579). The licensee affirmed the applicability of the following NSHC determination in its application dated September 26, 2003.

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 allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. Being in a TS condition and the associated required actions is not an initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. The consequences of an accident while relying on required actions as allowed by proposed LCO 3.0.4, are no different than the consequences of an accident while entering and relying on the required actions while starting in a condition of applicability of the TS. Therefore, the consequences of an accident previously evaluated are not significantly affected by this change. The addition of a requirement to assess and manage the risk introduced by this change 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). Entering into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS, will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously evaluated. The addition of a requirement to assess and manage the risk introduced by this change will further minimize possible concerns. Thus, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

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

The proposed change allows entry into a mode or other specified condition in the applicability of a TS, while in a TS condition statement and the associated required actions of the TS. The TS allow operation of the plant without the full complement of equipment through the conditions for not meeting the TS LCO. The risk associated with this allowance is managed by the imposition of required actions that must be performed within the prescribed completion times. The net effect of being in a TS condition on the margin of safety is not considered significant. The proposed change does not alter the required actions or completion times of the TS. The proposed change allows TS conditions to be entered, and the associated required actions and completion times to be used in new circumstances. This use is predicated upon the licensee's performance of a risk assessment and the management of plant risk. The change also eliminates current allowances for utilizing required actions and completion times in similar circumstances, without assessing and managing risk. The net change to the margin of safety is insignificant. Therefore, this change does not involve a significant reduction in a margin of safety.

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

Attorney for licensee: Richard F. Locke, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Section Chief: Stephen Dembek.

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

Date of amendment request: October 23, 2003.

Description of amendment request: The proposed amendment would delete the surveillance requirements associated with the Emergency Diesel Generator lockout features.

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 changes to the Technical Specifications (TS) 3/4.8.1.1, AC Sources—Operating, would delete an unnecessary surveillance. The probability of occurrence or the consequences for an accident or malfunction of equipment is not increased by the proposed changes. In addition, the proposed changes do not alter the way any structure, system or component (SSC) functions, do not modify the manner in which the plant is operated, and do not significantly alter equipment out-of-service time. Deleting the surveillance of equipment protection does not change the probability or consequences of any accident and dose consequences are unaffected. No changes to the design of structures, systems, or components (SSC) are made and there are no effects on accident mitigation.

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

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

Response: No.

The possibility of a new or different kind of accident from any accident or malfunction in the Hope Creek Updated Final Safety Analysis Report (UFSAR) is not created. The Emergency Diesel Generators are accident mitigation equipment and cannot initiate an accident. The proposed changes to the TS do not change the design function or operation of any SSCs. The TS, as amended, would continue to provide assurance of EDG operability.

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

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

Response: No.

The proposed changes are procedural in nature and make no changes that affect the ability of plant SSCs to perform their design basis accident functions. In addition, the proposed changes do not change the margin of safety since no SSCs are changed. The results of accident analysis remain unchanged by the proposed changes to TS.

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

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

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: October 24, 2003.

Description of amendment request: The proposed change to Technical Specifications will revise surveillance requirements associated with reactor protection system instrumentation, control rod block instrumentation, source range monitors, and power distribution limits, to minimize unnecessary testing.

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

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

Response: No.

The proposed amendment would revise the Technical Specification (TS) Surveillance Requirements (SRs) for certain Reactor Protection System and Control Rod Block Instrumentation, the source range monitors and power distribution limits, consistent with NUREG–1433, ''Standard Technical Specifications (STS) General Electric Plants, BWR [Boiling Water Reactor]/4,'' Revision 2. No changes are being made to any instrumentation setpoints or plant components. The revised SRs continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the Limiting Conditions for Operation will be met.

Since the proposed changes do not affect any accident initiator and since the associated equipment will remain capable of performing its design function, the proposed change does not involve a significant increase in the probability or radiological 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 changes do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

No changes are being made to any plant instrumentation setpoints or to the required level of redundancy. No changes are being made to any power distribution limits.

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

NRC Section Chief: James W. Clifford.

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

Date of amendment request: September 12, 2003.

Description of amendment request: The proposed amendments would revise Technical Specification (TS) Sections 1.1, 3.7.10, 3.7.12, 3.7.13, 3.7.14, 3.9.4, 5.5.2, and 5.5.10, and the associated Bases Sections to implement an alternate source term at North Anna Power Station, Units 1 and 2. The proposed changes would implement NUREG–1465, ''Accident Source Terms for Light-Water Nuclear Power Plants,'' dated February 1995, as the design-basis source term, achieve a consistent design basis for all accident dose assessments, increase operational flexibility by allowing for increased emergency core cooling system leakage and unfiltered control room in-leakage, and eliminate the surveillance requirement to test the bottled air flow rate.

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:

We have reviewed the proposed TS changes relative to the requirements of 10 CFR 50.92 and determined that a significant hazards consideration is not involved. Specifically, operation of North Anna Power Station with the proposed changes will not:

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

The proposed amendment does not involve a significant increase in the probability or consequence of an accident previously analyzed. The North Anna MCR/ESGR [main control room/emergency switchgear room] EVS [emergency ventilation system], PREACS [pump room exhaust air cleanup system], and MCE [MCR]/ESGR Bottled Air systems only function following the initiation of a design basis radiological accident. Therefore, the changes to these specifications, the definition of currently irradiated fuel, and the increase [of] the depressurization time of [the] containment following a design basis LOCA [loss-of-coolant accident] will not increase the probability of any previously analyzed accident. These systems are not initiators of any design bases accident.

Revised dose calculations, which take into account the changes proposed by this [these] amendment[s] and the use of the alternative source term[,] have been performed for the North Anna design basis radiological accidents. The results of these revised calculations indicate that public and control room doses will not exceed the limits specified in 10 CFR 50.67 and Regulatory Guide 1.183. There is not a significant increase in predicted dose consequences for any of the analyzed accidents. Therefore, the proposed changes do not involve a significant increase in the consequences of any previously analyzed accident.

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

The implementation of the proposed changes does not create the possibility of an accident of a different type than was previously evaluated in the UFSAR [Updated Final Safety Analysis Report]. Although the proposed changes could affect the operation of the MRC [MCR]/ESGR EVS following a design basis radiological accident, none of these changes can initiate a new or different kind of accident since they are only related to system capabilities that provide protection from accidents that have already occurred. These changes do not alter the nature of events postulated in the UFSAR nor do they introduce any unique precursor mechanisms.

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

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

The implementation of the proposed changes does not reduce the margin of safety. The proposed changes for the MCR/ESGR EVS, PREACS, and MCE [MCR]/ESGR Bottled Air System do not affect the ability of these systems to perform their intended safety functions to maintain dose less than the required limits during design basis radiological events. The revised dose calculations also indicate that the change to the containment depressurization times will continue to maintain the dose to the public and control room operators less than the required limits.

The radiological analysis results, when compared with the revised TEDE [total effective dose equivalent] acceptance criteria, meet the applicable limits. These acceptance criteria have been developed for application to analyses performed with alternative source terms. These acceptance criteria have been developed for the purpose of use in design basis accident analyses such that meeting the stated limits demonstrates adequate protection of public health and safety. It is thus concluded that the margin of safety will not be reduced by the implementation of the changes.

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: Ms. Lillian M. Cuoco, Esq., Senior Counsel, Dominion Resources Services, Inc., Millstone Power Station, Building 475, 5th Floor, Rope Ferry Road, Rt. 156, Waterford, Connecticut 06385.

NRC Section Chief: John A. Nakoski.

Notice of Issuance of Amendments to Facility Operating Licenses

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

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

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

For further details with respect to the action see (1) The applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, Public File Area 01F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, 301–415–4737 or by e-mail to *pdr@nrc.gov.*

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

Date of application for amendment: June 2, 2003.

Brief description of amendment: The amendment revised the Technical Specifications, Sections 3.7.B.1 and 3.7.C.2. Section 3.7.B.1 required that the reactor may remain in operation ''for a period not to exceed 7 days in any 30 day period if a startup transformer is out of service.'' Section 3.7.C.2 required that the reactor may be in operation ''for a period not to exceed 7 days in any 30 day period if a diesel generator is out of service.'' The amendment deleted the phrase ''in any 30 day period'' from these two sections.

Date of Issuance: November 24, 2003.

Effective date: November 24, 2003 and shall be implemented within 30 days of issuance.

Amendment No.: 239.

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

Date of initial notice in **Federal Register:** July 8, 2003 (68 FR 40709).

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated November 24, 2003.

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: March 27, 2002, as supplemented on May 30, July 10, October 10, October 28, November 26, and December 18, 2002, and on January 6, January 27, February 26, April 8, May 19, June 23, June 26, July 15, August 6, September 11, October 8, and October 14, 2003.

Brief description of amendment: This amendment converts the current Technical Specifications (TS) to a set of Improved TS based on NUREG–1431, Revision 2, ''Standard Technical Specifications for Westinghouse Plants,'' Revision 2, dated June 2001.

Date of issuance: November 21, 2003.

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

Amendment No.: 238.

Facility Operating License No. DPR–26: Amendment replaced the current Technical Specifications (TSs) with the Improved TSs in their entirety and revised the license.

Date of initial notice in **Federal Register:** *September 26, 2003 (68 FR 55660).*

The supplemental letters that were received subsequent to the issuance of the **Federal Register** notice provided clarifying information that did not change the no significant hazards consideration.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 21, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: March 19, 2003.

Brief description of amendment: This amendment deletes Technical Specification (TS) 5.5.3, ''Post Accident Sampling,'' and thereby eliminates the requirements to have and maintain the post accident sampling system at the Pilgrim Nuclear Power Station.

Date of issuance: November 14, 2003.

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

Amendment No.: 204.

Facility Operating License No. DPR–35: Amendment revised the TSs.

Date of initial notice in **Federal Register:** *June 10, 2003 (68 FR 34663).*

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 2003.

No significant hazards consideration comments received: No.

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

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

Date of application for amendments: October 16, 2002, as supplemented by letters dated June 20, 2003 and October 14, 2003.

Brief description of amendments: The amendments revise the completion time of Required Action A.1 of Technical Specification 3.8.7, ''Inverters-Operating,'' from the current 24 hours to 7 days for one inoperable instrument bus inverter. This provides greater operational flexibility for online maintenance of an instrument bus inverter with the potential to reduce the duration of refueling outages.

Date of issuance: November 19, 2003.

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

Amendment Nos.: 135/135, 129/129.

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

Date of initial notice in **Federal Register:** December 10, 2002 (67 FR 75874).

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

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 2003. 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: October 24, 2002 and as supplemented by letter dated June 20, 2003.

Brief description of amendments: The amendments revise Technical Specification 5.5.13, ''Primary Containment Leakage Rate Testing Program,'' to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009 for Unit 1 and no later than December 7, 2008 for Unit 2.

Date of issuance: November 19, 2003.

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

Amendment Nos.: 162, 148.

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

Date of initial notice in **Federal Register:** December 10, 2002 (67 FR 75876).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 19, 2003.

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: May 19, 2003.

Brief description of amendments: The amendments revise Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF–11 and NPF–18. Specifically, the change will decrease the frequency associated with TS Surveillance Requirement (SR) 3.7.7.1 for Turbine Bypass Valve (BPV) testing from 7 to 31 days. The change is consistent with the testing frequency contained in NUREG–1434, ''Standard Technical Specifications General Electric Plants, BWR/6,'' Revision 2, dated June 2001, for BPV testing. The 7-day frequency associated with SR 3.7.7.1 was established in the LaSalle County Station (LSCS) TS during conversion to improved Standard Technical Specifications (STS) format due to the testing frequency contained in the LSCS custom TS and the difficulties experienced with other Electro-Hydraulic Control (EHC) system valves to consistently pass their surveillance tests. LSCS has recently re-evaluated the performance of these valves and has determined that the current performance of these valves supports decreasing the testing frequency of the BPVs from 7 to 31 days.

Date of issuance: November 13, 2003.

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

Amendment Nos.: 163/148.

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

Date of initial notice in **Federal Register:** June 24, 2003 (68 FR 37577).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 13, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 19, 2002, as supplemented July 25, 2003.

Brief description of amendment: The proposed amendment would revise the Kewaunee technical specifications to change the Nuclear Regulatory Commission reporting requirements for the discovery of defective or degraded steam generator tubes so that the requirements are aligned with 10 CFR 50.72 and 10 CFR 50.73.

Date of issuance: November 20, 2003.

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

Amendment No.: 171.

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

Date of initial notice in **Federal Register:** January 21, 2003 (68 FR 2807).

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 amendment is contained in a Safety Evaluation dated November 20, 2003.

No significant hazards consideration comments received: No.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama Southern Nuclear Operating Company, Inc., et al., Docket Nos. 50–424 and 50–425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of amendments request: September 2, 2003.

Brief Description of amendments: The amendments extend from 1 hour to 24 hours the completion time for Condition B of Technical Specification 3.5.1, which defines requirements for the restoration of an emergency core cooling system accumulator when it has been declared inoperable for a reason other than boron concentration.

Date of issuance: November 18, 2003.

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

Amendment Nos.: 162, 155, 129, & 107.

Facility Operating License Nos. NPF–2 and NPF–8: Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** October 14, 2003 (68 FR 59220).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 18, 2003.

No significant hazards consideration comments received: No.

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

Date of application for amendment: December 19, 2002.

Brief description of amendment: The amendment consists of changes to Technical Specification (TS) 5.0, ''Administrative Controls,'' to incorporate three approved TS Task Force (TSTF) changes: TSTF–258, Revision 4, ''Changes to Section 5.0, Administrative Controls''; TSTF–299, Revision 0, ''Administrative Controls Program 5.5.2.b Test Interval and Exception''; and TSTF–308, Revision 1, ''Determination of Cumulative and Projected Dose Contributions in the Radioactive Effluent Controls Program.'' In addition, two editorial changes are incorporated to update personnel titles and clarify required staffing levels.

Date of issuance: November 13, 2003.

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

Amendment No.: 49.

Facility Operating License No. NPF–90: Amendment revised the TSs.

Date of initial notice in **Federal Register:** April 1, 2003 (68 FR 15764).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 13, 2003.

No significant hazards consideration comments received: No.

For the Nuclear Regulatory Commission

Dated at Rockville, Maryland, this 1st day of December 2003.

Ledyard B. Marsh,

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

[FR Doc. 03–30246 Filed 12–8–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Regulatory Guide; Issuance, Availability

The Nuclear Regulatory Commission (NRC) has issued a new guide in its Regulatory Guide Series. This series has been developed to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in its review of applications for permits and licenses, and data needed by the NRC staff in its review of applications for permits and licenses.

Regulatory Guide 1.199, ''Anchoring Components and Structural Supports in Concrete,'' has been developed to provide guidance to licensees and applicants on methods acceptable to the NRC staff for complying with the NRC's regulations in the design, evaluation, and quality assurance of anchors (steel embedments) used for component and structural supports on concrete structures.

Comments and suggestions in connection with items for inclusion in guides currently being developed or improvements in all published guides are encouraged at any time. Written comments may be submitted to the Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Questions on the content of this guide may be directed to Mr. H. Graves, (301) 415–5880; *e-mail hlg1@nrc.gov.*

Regulatory guides are available for inspection or downloading at the NRC's Web site at *http://www.nrc.gov* under Regulatory Guides and in NRC's Electronic Reading Room (ADAMS System) at the same site. Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301) 415–2289, or by e-mail to *distribution@nrc.gov.* Issued guides may also be purchased from the National Technical Information Service (NTIS) on a standing order basis. Details on this service may be obtained by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; telephone 1–800–553–6847; *http://www.ntis.gov.* Regulatory guides are not copyrighted, and Commission approval is not required to reproduce them.

(5 U.S.C. 552(a))

Dated at Rockville, MD this 28th day of November 2003.

For the Nuclear Regulatory Commission.

Ashok C. Thadani,

Director, Office of Nuclear Regulatory Research.

[FR Doc. 03–30467 Filed 12–8–03; 8:45 am]

BILLING CODE 7590–01–P

POSTAL RATE COMMISSION

Facility Tour

AGENCY: Postal Rate Commission.

ACTION: Notice of Commission tour.

SUMMARY: Postal Rate Commissioners and several staff members will tour United Parcel Service (UPS) facilities on December 11 and 12, 2003. On the evening of December 11, from approximately 8 p.m. to 10 p.m., the group will tour the UPS Mail Innovations facility in Paulsboro, NJ. On December 12, from approximately 11:30 a.m. to 1:15 p.m., the group will tour the UPS air hub at the Philadelphia airport. The purpose of the tours (including any related briefings) is to observe operations.

DATES: (1) December 11, 2003: UPS facilities (Paulsboro, NJ). (2) December 12, 2003; UPS facilities (Philadelphia Airport Hub).

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

Dated: December 4, 2003.

Garry J. Sikora,

Acting Secretary.

[FR Doc. 03–30434 Filed 12–8–03; 8:45 am]

BILLING CODE 7710–FW–M

SECURITIES AND EXCHANGE COMMISSION

Submission for OMB Review; Comment Request

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

Extension: Rule 11Ac1–1; SEC File No. 270–404; OMB Control No. 3235–0461.

Notice is hereby given that pursuant to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*), the Securities and Exchange Commission (''Commission'') has submitted to the Office of Management and Budget (''OMB'') a request for extension of the previously approved collection of information discussed below.

Rule 11Ac1–1, Dissemination of Quotations, contains two related collections of information necessary to