[7590-01-P]

NUCLEAR REGULATORY COMMISSION

# BIWEEKLY NOTICE

# APPLICATIONS AND AMENDMENTS TO FACILITY OPERATING LICENSES INVOLVING NO SIGNIFICANT HAZARDS CONSIDERATIONS

[NRC-2010-0256]

## I. Background

Pursuant to Section 189a.(2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (the Commission or NRC) is publishing this regular biweekly notice. The Act requires the Commission publish notice of any amendments issued, or proposed to be issued and 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 July 1, 2010 to July 14, 2010. The last biweekly notice was published on July 13, 2010 (75 FR 39975).

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 Title 10 of the *Code of Federal Regulations* (10 CFR), Section 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 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the *Federal Register* a notice of issuance. Should the Commission make a final No Significant

Hazards Consideration Determination, any hearing will take place 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, Announcements, and Directives Branch (RADB), TWB-05-B01M, 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 faxed to the RADB at 301-492-3446. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, Public File Area O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license. 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 person(s) should consult a current copy of 10 CFR 2.309, which is available at the Commission's PDR, located at One White Flint North, Public File Area O1F21, 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 a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, 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 general requirements: 1) the name, address, and telephone number of the requestor or petitioner; 2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; 3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and 4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the requestor/petitioner seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the requestor/petitioner shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner intends to rely to establish those facts or expert opinion. The petition must include 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 requestor/petitioner to relief. A requestor/petitioner who fails to satisfy 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.

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.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least ten (10) days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and

(2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through EIE, users will be required to install a Web browser plug-in from the NRC Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than

11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants.

Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket which is available to the public at http://ehd.nrc.gov/EHD_Proceeding/home.asp, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Non-timely filings will not be entertained absent a determination by the presiding officer that the petition or request should be granted or the contentions should be admitted, based on a balancing of the factors specified in 10 CFR 2.309(c)(1)(i)–(viii).

For further details with respect to this license amendment application, 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 O1F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

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

Date of amendment request: April 8, 2010.

Description of amendment request: The amendments would revise Technical Specification (TS) 2.2, "Safety Limit Violations," consistent with Technical Specification Task Force (TSTF) change traveler TSTF-5-A, and TS 5.2.1, "Onsite and Offsite Organizations," consistent with TSTF-65-A, Revision 1. Specifically, the proposed amendment would delete redundant reporting and operational restriction provisions from TS 2.2 and replace plant-specific organization titles with generic organization titles in TS 5.2.1. Both TSTF-5-A and TSTF-65-A were incorporated in Revision 2 of NUREG-1432, "Standard Technical Specifications for Combustion Engineering Plants."

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

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

   Response: No.

   These changes involve minor changes in organization titles and remove redundant and unnecessary reporting requirements. The changes are consistent with TSTF-5 and TSTF-65, which have been approved by the NRC Staff, and included in Revision 2 of NUREG-1432. Technical Specification Safety Limit violation reporting is redundant to 10 CFR 50.36(c)(7) and (8) and 10 CFR 50.72 and 73. The removal of the notification, reporting, and startup requirements from the TS is an administrative change because the current requirements duplicate what is already contained in the regulations. The proposed changes do not alter existing controls on plant operation (i.e., safety limit values, LCOs [Limiting Conditions for Operations], Surveillance Requirements or Design Features), but only remove the administrative burden of maintaining redundant notification, reporting, and plant startup requirements.

Functions which are necessary to operate the facility safely and in accordance with the operating licenses remain within the organization and will not affect the safe operation of the plant and will continue to ensure proper control of administrative activities. The proposed changes will not affect the operation of structures, systems, or components, and will not reduce programmatic controls such that plant safety would be affected.

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

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

   Response: No.

   The proposed changes will not affect the operation of structures, systems, or components, and will not reduce programmatic controls such that plant safety would be affected. The generic title changes and deletion of redundant reporting are administrative.

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

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

   Response: No.

   The changes are administrative and will not diminish any organizational or administrative controls currently in place. The proposed change will not affect the operation of structures, systems, or components, and will not reduce programmatic controls such that plant safety would be affected. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on 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: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072-2034.

NRC Branch Chief: Michael T. Markley.

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: April 29, 2010.

Description of amendment request: The amendments would revise the Technical Specifications (TSs) to incorporate Technical Specifications Task Force (TSTF) change traveler TSTF-479-A, “Changes to Reflect Revision of 10 CFR 50.55a,” as modified by TSTF-497-A, “Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less.” Specifically, the changes associated with TSTF-479-A would modify the reference in TS 5.5.8, “Inservice Testing Program,” from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) and would specify that the extension allowance of SRs is applicable to the frequencies in the Inservice Testing Program (IST). The changes associated with TSTF-497-A would limit the applicability of SR 3.0.2 to frequencies of 2 years or less. In addition, the amendment would remove the reference to component supports for consistency with the

Standard Technical Specifications because the supports are included in the licensee's Inservice Inspection Program.

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

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

   Response: No.

   The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the IST of pumps and valves and eliminates a statement regarding the testing of supports. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and will align the PVNGS specification wording to that of NUREG-1432, Revision 3.1, Standard Technical Specifications Combustion Engineering Plants. The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events, nor does it involve the addition or removal of any equipment, or any design changes to the facility.

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

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

   Response: No.

   The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the IST of pumps and valves and eliminates a statement regarding the testing of supports. The proposed change incorporates revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and aligns wording to that of the standard specification.

   The proposed changes do not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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

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

Response: No.

The proposed changes revise TS 5.5.8, "Inservice Testing Program," for consistency with the requirements of 10 CFR 50.55a(f)(4) regarding the inservice testing of pumps and valves and eliminates a statement regarding the testing of supports. The proposed changes incorporate revisions to the ASME Code that result in a net improvement in the measures for testing pumps and valves and the editorial change eliminates confusion as to the testing program for supports and aligns wording to that of the standard specification. The safety functions of the affected pumps and valves 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 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.

<ins>Attorney for licensee</ins>: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072-2034

<ins>NRC Branch Chief</ins>: Michael T. Markley.

<ins>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</ins>

<ins>Date of amendment request</ins>: April 29, 2010.

Description of amendment request: The amendments would remove the Main Steam and Main Feedwater Valve Isolation Times from the Technical Specifications (TSs) in accordance with Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) Standard Technical Specification change traveler TSTF-491, Revision 2, “Removal of the Main Steam and Main Feedwater Valve Isolation Times from Technical Specifications.” The isolation times would be located outside of the TSs in a document subject to control by the 10 CFR 50.59 process.

The NRC staff issued a Notice of Availability of “Technical Specification Improvement to Remove the Main Steam and Main Feedwater Valve Isolation Time from Technical Specifications Using the Consolidated Line Item Improvement Process,” associated with TSTF-491, Revision 2, in the *Federal Register* on December 29, 2006 (71 FR 78472). The notice included a model license amendment request. The notice also announced that the previously published (71 FR 193, October 5, 2006) model safety evaluation and model No Significant Hazards Consideration (NSHC) determination may be referenced in plant-specific applications to adopt the changes. In its application dated April 29, 2010, the licensee affirmed the applicability of the model NSHC determination which is presented below.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of NSHC adopted by the licensee 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 relocating main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. The proposed change is described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF-491 related to relocating the main steam and main feedwater valves isolation times to the Licensee Controlled Document that is referenced in the Bases and replacing the isolation time with the phase, "within limits."
>
> The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes relocate the main steam and main feedwater isolation valve times to the Licensee Controlled Document that is referenced in the Bases. The requirements to

perform the testing of these isolation valves are retained in the TS. Future changes to the Bases or licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, " Changes, test and experiments", to ensure that such changes do not result in more than minimal increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures.

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

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

The proposed changes relocate the main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the phase "within limits". The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal [plant] operation. The requirements in the TS continue to require testing of the main steam and main feedwater isolation valves to ensure the proper functioning of these isolation valves.

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

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

The proposed changes relocate the main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the phase "within limits". Instituting the proposed changes will continue to ensure the testing of main steam and main feedwater isolation valves. Changes to the Bases or license controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that main steam and feedwater isolation valve testing is conducted such that there is no significant reduction in the margin of safety.

The margin of safety provided by the isolation valves is unaffected by the proposed changes since there continue to be TS requirements to ensure the testing of main steam and main feedwater isolation valves. The proposed changes maintain sufficient controls to preserve the current margins of safety.

The NRC staff has reviewed the analysis adopted by the licensee 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 NSHC.

Attorney for licensee: Michael G. Green, Senior Regulatory Counsel, Pinnacle West Capital Corporation, P.O. Box 52034, Mail Station 8695, Phoenix, Arizona 85072-2034.

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: June 17, 2010.

Description of amendment request: The proposed change would revise Technical Specification (TS) 6.5.16, “Containment Leakage Rate Testing Program,” to allow for the extension of the 10-year frequency of the Arkansas Nuclear One, Unit 2 (ANO-2) Type A or Integrated Leak Rate Test (ILRT) to be extended to 15 years on a permanent basis.

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 involves changes to the ANO-2 Containment Leakage Rate Testing Program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators.

Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment adopts the NRC-accepted guidelines of [Nuclear Energy Institute (NEI)] 94-01, Revision 2-A ["Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008], for development of the ANO-2 performance-based testing program. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its components will limit leakage rates to less the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval to 15 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem [roentgen equivalent man] per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in [NRC Regulatory Guide] 1.174 ["An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"]. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. ANO-2 has determined that the increase in Conditional Containment Failure Probability due to the proposed change would be very small.

Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that 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 amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the ANO-2 performance-based leakage testing program, and establishes a 15-year interval for the performance of the containment ILRT. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled.

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

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

Response: No.

The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 2-A, for the development of the ANO-2 performance-based leakage testing program, and establishes a 15 year interval for the performance of the containment ILRT. This amendment does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the Containment Leakage Rate Testing Program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests will be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 2-A.

Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current ANO-2 PSA [Probabilistic Safety Assessment] model concluded that extending the ILRT test interval from ten years to 15 years results in a very small change to the ANO-2 risk profile.

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: Joseph A. Aluise, Associate General Counsel – Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

NRC Branch Chief: Michael T. Markley.

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 request: June 14, 2010.

Description of amendment request: The proposed amendments would allow a revision of the licensing basis, as described in the Final Safety Analysis Report Update (FSARU), to include damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. In addition, the RG 1.61, Revision 1, Table 1 note allowing the use of a "weighted average" for design-basis safe-shutdown earthquake (SSE) damping values applicable to steel structures of different connection types will also be applied to determine the IHA design-basis operating-basis earthquake (OBE) damping values.

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

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

   The proposed change would allow use of critical damping values consistent with the recommendations of RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, dated March 2007, for the seismic design and analysis of the IHA. The RG 1.61, Revision 1, Table 1 note allowing use of a "weighted average" for design-basis SSE damping values applicable to steel structures of different connection types, is also applied to determine the IHA design-basis OBE damping values. RG 1.61, Revision 1, Table 2 for OBE damping values does not contain the same note as found in Table 1. However use of the note for the determination of the DE [design earthquake] damping value is consistent with the use of the note for the determination of the DDE [double design earthquake] and HE [Hosgri earthquake] damping values, and a weighted average more realistically represents the IHA structure.

RG 1.61, Revision 1, specifies the damping values that the NRC staff currently considers acceptable for complying with the agency's regulations and guidance for seismic analysis. Revision 1 incorporates the latest data and information, and reduces unnecessary conservatism in specification of damping values for seismic design and analysis of SSCs [structures, systems, and components].

The proposed change does not change the design functions of the IHA or its response to design-basis events, nor does it affect the capability of related SSCs to perform their design or safety functions. The use of the proposed damping values in the seismic design and analysis of the IHA is related to the ability of the IHA to function in response to design-basis seismic events, and is unrelated to the probability of occurrence of those events, or other previously evaluated accidents. Therefore the proposed change will not have any impact on the probability of an accident previously evaluated.

The proposed damping values are an element of the seismic analyses performed to confirm the ability of the IHA to function under postulated seismic events while maintaining resulting stresses within ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] Section III allowable values. Therefore, the use of damping values consistent with the recommendations of RG 1.61, Revision 1 does not result in an increase in the consequences of accidents previously evaluated.

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

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

   The proposed change does not involve changes to any plant SSCs, nor does it involve changes to any plant operating practice or procedure. The damping values are an element of the seismic analyses performed to confirm the ability of the IHA to function under postulated seismic events while maintaining resulting stresses within ASME Section III allowable values. Therefore, no credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases are created that would create the possibility of a new or different kind of accident.

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 change involve a significant reduction in a margin of safety?

The design basis of the plant requires structures to be capable of withstanding normal and accident loads including those from a design basis earthquake. The proposed change would allow the use of damping values in the IHA seismic analyses that are in general more realistic and, thus, more accurate than the damping values recommended in RG 1.61, Revision 0, used in the analysis for the HE, or the plant specific damping values used in the original analysis for the DE and DDE. The NRC stated, in NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Supplement No. 7, that allowing use of the higher damping values in RG 1.61, Revision 0 for the HE re-evaluation, versus the lower values used in the original analysis, is realistic and should not be regarded as an arbitrary lowering of the margins of safety. The damping values in RG 1.61, Revision 0, were based on limited data, expert opinion, and other information available in 1973. NRC and industry research since 1973 show that the damping values provided in the original version of RG 1.61 may not reflect realistic damping values for SSCs. RG 1.61, Revision 1, therefore, provides damping values based on the updated research results that predict and estimate damping values for seismic design of SSCs in nuclear power plants, and similarly should not be regarded as an arbitrary lowering of the margins of safety.

As discussed above, damping values are an element of the seismic analyses performed to confirm the ability of the IHA to function during design-basis seismic events while maintaining resulting stresses within ASME Section III allowable values. The proposed change [to] allow use of damping values consistent with the recommendations of RG 1.61, Revision 1, versus the damping values in the current licensing basis could result in lower calculated stresses. The analysis done for the IHA using the proposed damping values showed the ASME Section III allowable values are met. Sufficient safety margins are maintained when Codes and standards or alternatives approved for use by the NRC are met.

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

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

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

NRC Branch Chief: Michael T. Markley.

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

Date of amendment request: May 28, 2010.

Description of amendment request: To revise Technical Specification (TS) 4.2.2 "Control Rod Assemblies." The proposed change would include silver-indium-cadmium material in addition to the boron carbide control rod material.

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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Watts Bar Unit 1 Technical Specification 4.2.2, Control Rod Assemblies, is revised to include [silver-indium-cadmium] Ag-In-Cd material in addition to the [boron carbide] B4C control rod material. In addition to the absorber material change, the replacement [enhanced performance] EP Ag-In-Cd [rod cluster control assemblies] RCCAs will be coupled with Control Rod Drive Mechanism (CRDM) drive rod shafts which are lighter than the CRDM drive rod shaft coupled to the B4C drive rod shafts. Also, the EP Ag-In-Cd RCCAs are heavier than the B4C RCCAs and have a different reactivity, or rod worth.

   There are a number of events that are related to inadvertent movement of the RCCAs; however, they are not initiated by the RCCAs. They are initiated by the failure of plant structures, systems, or components (SSC) other than the RCCAs. The proposed changes to the RCCA design do not have a detrimental impact on the integrity of any plant SSC that initiates an analyzed event. In addition, the EP Ag-In-Cd RCCAs have the capability to mitigate events, because:

a) The Ag-In-Cd RCCA/standard drive line weight continues to meet the rod drop time of 2.7 seconds limit listed in Technical Specification 3.1.5 (Rod Group Alignment Limits); and

b) The reactivity difference was addressed for the impact on core neutronics and safety analyses. It was determined that the reactivity change can be accommodated within the bounds of the current safety analysis limits using approved NRC methodology. Future core designs will use an NRC approved methodology as the means to demonstrate the continued safe operation of the plant with the EP Ag-In-Cd RCCAs.

The change does not adversely affect the protective and mitigative capabilities of the plant, nor does the change affect the initiation or probability of occurrence of any accident. The SSCs will continue to perform their intented safety functions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   Watts Bar Unit 1 Technical Specification 4.2.2, Control Rod Assemblies, is revised to include Ag-In-Cd material in addition to the B4C control rod material. In addition to the absorber material change, the replacement EP Ag-In-Cd RCCAs will be coupled with Control Rod Drive Mechanism (CRDM) drive rod shafts which are lighter than the CRDM drive rod shaft coupled to the B4C drive rod shafts. Also, the EP Ag-In-Cd RCCAs are heavier than the B4C RCCAs and have a different reactivity, or rod worth.

   The EP Ag-In-Cd RCCAs are identical to the current RCCAs in terms of form, fit, and function. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis. The possibility of a new or different malfunction of safety-related equipment is not created. No new accident scenarios, transient precursors, or limiting single failures are introduced as a result of these changes. There will be no adverse effects or challenges imposed on any safety-related system as a result of these changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No.

   Watts Bar Unit 1 Technical Specification 4.2.2, Control. Rod Assemblies, is revised to include Ag-In-Cd material in addition to the B4C control rod material. In addition to the absorber material change, the replacement EP Ag-In-Cd RCCAs will be coupled with Control Rod Drive Mechanism (CRDM) drive rod shafts which are lighter than the CRDM drive rod shaft coupled to the B4C drive rod shafts. Also, the EP Ag-In-Cd RCCAs are heavier than the B4C RCCAs and have a different reactivity, or rod worth. The changes in weight and reactivity of the CRDM/RCCA on the design criteria and safety analysis have been addressed.

The proposed changes regarding the Ag-In-Cd RCCAs do not involve a significant reduction in a margin of safety, because:

a) The Ag-In-Cd RCCA/standard drive line weight continues to meet the rod drop time of 2.7 seconds limit listed in Technical Specification 3.1.5 (Rod Group Alignment Limits); and

b) The reactivity difference was addressed for the impact on core neutronics and safety analyses. It was determined that the reactivity change can be accommodated within the bounds of the current safety analysis limits using approved NRC methodology. Future core designs will use an NRC approved methodology as the means to demonstrate the continued safe operation of the plant with the EP Ag-In-Cd RCCAs.

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: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Stephen J. Campbell.

## 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 51.22(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 (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 (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 PDR Reference staff at 1 (800) 397-4209, (301) 415-4737 or by email to pdr.resource@nrc.gov.

FirstEnergy Nuclear Operating Company, et al., Docket No. 50-334

Beaver Valley Power Station, Unit No. 1 (BVPS-1), Beaver County, Pennsylvania

Date of application for amendment: July 6, 2009, as supplemented on March 10, 2010

Brief description of amendment: The amendment revises Technical Specification (TS) 5.6.3, "Core Operating Limits Report," to allow the use of the generically approved Topical Report, WCAP-16009-P-A, "Realistic Large Break LOCA [Loss-of-Coolant Accident] Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method," for BVPS-1.

Date of issuance: July 1, 2010

Effective date: As of the date of issuance, and shall be implemented prior to startup following the fall 2010 maintenance and refueling outage.

Amendment No: 286

Facility Operating License No. DPR-66: The amendment revised the License and TS.

Date of initial notice in FEDERAL REGISTER: December 1, 2009 (74 FR 62835).

The March 8, 2010, supplement provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 1, 2010.

No significant hazards consideration comments received: No

Nine Mile Point Nuclear Station, LLC, Docket No. 50-220, Nine Mile Point Nuclear Station, Unit No. 1 (NMP1), Oswego County, New York

Date of application for amendment: July 2, 2009.

Brief description of amendment:  The amendment revises the TSs by removing position indication for the relief valves and safety valves from TS 3.6.11, “Accident Monitoring Instrumentation.”  The amendment would also correct an editorial error in the title of Table 4.6.11, “Accident Monitoring Instrumentation Surveillance Requirement.”

Date of issuance:  June 29, 2010.

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

Amendment No.:  205.

Renewed Facility Operating License No. DPR-63:  The amendment revises the License and TSs.

Date of initial notice in FEDERAL REGISTER:  October 14, 2009 (74 FR 52826).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated June 29, 2010.

No significant hazards consideration comments received:  No.

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

Date of application for amendment: November 30, 2009.

Brief description of amendment: The amendment revises the emergency diesel generator (DG) Completion Time for inoperable DGs in Technical Specification (TS) 3.8.1, “AC Sources Operating.” The amendment revises the Completion Time from 14 days to 72 hours for restoring one or more inoperable DG(s) in one train to an operable status. The amendment was requested because of the potential completion and startup of the WBN Unit 2.

Date of issuance: July 6, 2010.

Effective date: As of the date of issuance and shall be implemented after the issuance of the facility operating license for WBN Unit 2 and prior to WBN Unit 2 entry into Mode 4, “Hot Shutdown.”

Amendment No.: 84.

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

Date of initial notice in *Federal Register*: March 9, 2010 (75 FR 10830).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 6, 2010.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 15th day of July 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Joseph G. Giitter, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation