APS

Palo Verde Nuclear
Generating Station

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10 CFR 50.90
10 CFR 50.55a(g)

102-05008-CDM/SAB/RKR
October 7, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2 and 3
Docket Nos. STN 50-528, 50-529, and 50-530
Request for Amendment to Technical Specifications 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program" and 5.5.16, "Containment Leakage Rate Testing Program"

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) hereby requests an amendment to Technical Specifications (TS) Section 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program" and Section 5.5.16, "Containment Leakage Rate Testing Program." The proposed change is consistent with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix).

The License Amendment Request (LAR) is similar to changes approved by the NRC for Oconee Nuclear Station, Units 1, 2 and 3 on January 18, 2000, Calvert Cliffs Nuclear Power Plant, Units 1 and 2 on January 30, 2001, and Vogtle Electric Generating Plant, Units 1 and 2 on June 6, 2001.

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Arizona Public Service Company (APS) is submitting this LAR in conjunction with an industry consortium of six stations as a result of a mutual agreement known as Strategic Teaming and Resource Sharing (STARS). The STARS group consists of the six stations operated by TXU Generation Company LP, Union Electric Company, Wolf Creek Nuclear Operating Corporation, Pacific Gas and Electric Company, STP Nuclear Operating Company, and Arizona Public Service Company. Other members of the group are expected to submit license amendment requests similar to this one. Due to differences between the STARS plants, there may be some differences in the plant LARs, particularly for the information provided in Enclosure 1.

Based on the responses to the three criteria provided for determining whether a significant hazard consideration exists as stated in 10 CFR 50.92, APS has concluded that the proposed amendment involves no significant hazard consideration.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter, this submittal is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10CFR 50.91(b)(1).

The proposed amendment changes the requirements for the "Pre-Stressed Concrete Containment Tendon Surveillance Program" (TS Section 5.5.6) and the "Containment Leakage Rate Testing Program" (TS Section 5.5.16). Since the requirements for these programs are being revised, APS requests that the following condition be added to the amendment issuance letter: "For surveillance requirements associated with the revised "Pre-Stressed Concrete Containment Tendon Surveillance Program" and the "Containment Leakage Rate Testing Program" in technical specifications 5.5.6 and 5.5.16, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the date of implementation of this amendment." This is consistent with the license condition issued with technical specification amendment 117 to the Palo Verde operating license.

The changes proposed in this LAR are not required to address an immediate safety concern. APS requests that this amendment be approved by May 1, 2004. Approval of this amendment would allow rescheduling of the next performance of the tendon surveillances from the summer, 2004 to 2008. APS requests the LAR be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

No commitments are being made to the NRC by this letter. If you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

CDM/SAB/RKR/kg

Enclosures:
- Notarized affidavit
- Licensee's evaluation of the proposed change(s)

Attachments:
1. Proposed Technical Specification Changes (mark-up)
2. Proposed Technical Specification pages (retyped)
3. Changes to TS Bases pages

cc:	B. S. Mallett	NRC Region IV Regional Administrator
	M. B. Fields	NRC NRR Project Manager
	N. L. Salgado	NRC Senior Resident Inspector for PVNGS
	J. N. Donohew	NRC
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)

ENCLOSURE 1

AFFIDAVIT

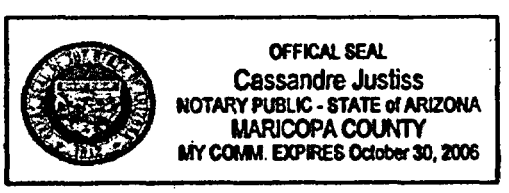
STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

Sworn To Before Me This 7th Day Of October, 2003.

Notary Public



Notary Commission Stamp

ENCLOSURE 2

ARIZONA PUBLIC SERVICE COMPANY'S EVALUATION

Subject: Request for Amendment to Technical Specifications 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program" and 5.5.16, "Containment Leakage Rate Testing Program"

1.0 DESCRIPTION

This letter is a request from Arizona Public Service Company (APS) to amend Operating Licenses NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, respectively.

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. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

2.0 PROPOSED CHANGE

The proposed change will revise:

- Technical Specification 5.5.6

  This specification is revised to replace the reference to Regulatory Guide 1.35 with the ASME Code Section XI as follows.

  "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR." is replaced with "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Code Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC." Additionally, the provisions of SR 3.0.2 are deleted from this specification.

- Technical Specification 5.5.16

  This specification is revised to add the following exception to Regulatory Guide 1.163, "Performance- Based Containment Leak-Testing Program,"

  "1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWL, except where relief has been authorized by the NRC. The containment concrete visual examinations may be performed during either power operation, e.g., performed

concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWE, except where relief has been authorized by the NRC."

In addition, the paragraphs in section 5.5.6 have been sequenced to more clearly separate the requirements of the program. This is considered an administrative change and is consistent with the guidance in NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," revision 2.

The TS Bases for SR 3.6.1.1 and SR 3.6.1.2 are also revised to indicate that the testing and frequency requirements of the Containment Tendon Surveillance Program are in accordance with ASME Code Section XI, Subsection IWL and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC. The markup of the proposed Bases change is provided in Attachment 3, for information only.

3.0 BACKGROUND

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed change to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code. The final rule, Subpart 50.55a(g)(6)(ii)(B) of Title 10 of the Code of Federal Regulations (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001. APS has implemented these requirements. This amendment request updates the TS to be consistent with the rule change.

The containment consists of a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome. The Post-tensioning System used for the shell and dome of the containment employs tendons. Each tendon consists of high strength steel wires and anchoring components. The tendons are installed in tendon ducts and tensioned in a predetermined sequence. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

4.0 TECHNICAL ANALYSIS

Technical Specification 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," states in part, "The Tendon Surveillance Program, inspection

frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR." As identified above, 10 CFR 50.55a(g)(4) requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix). The requirements in 10 CFR 50.55a(g)(4) and ASME Code Section XI, Subsection IWL do not reference Regulatory Guide 1.35, Revision 1. As such, the TS are inconsistent with the requirements of 10 CFR 50.55a.

10 CFR 50.55a(g)(5)(ii) states, in part: "If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for Amendment of the technical specifications to conform the technical specification to the revised program." Based on the requirements in 10 CFR 50.55a, PVNGS is required to update the Technical Specifications. The containment inservice inspection programs are required to be in accordance with ASME Code Section XI, Subsection IWL as modified by 10 CFR 50.55a(b)(2)(viii), except where an exemption or relief has been authorized by the NRC.

Additionally, since the tendon inspection frequencies will be in accordance with ASME Code Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable and are deleted from Technical Specification 5.5.6. 10 CFR 50.55a requires the implementation of ASME Code Section XI, Subsection IWL and specifies the requirements for extending inspection frequencies.

Technical Specification 5.5.16 contains requirements for the Containment Leakage Rate Testing Program, and it specifies that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Position C.3 of the regulatory guide states that, "Section 9.2.1, "Pretest Inspection and Test Methodology," of NEI 94-01 provides guidance for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of Regulatory Guide 1.163 and NEI 94-01, the concrete surfaces of the containment must be visually examined in accordance with the ASME Code Section XI, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every five years, and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The visual examinations performed pursuant to Subsection IWL may be performed at any time during power operation or

during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this in the only time that the liner plate is fully accessible.

In addition, the visual examinations performed pursuant to Subsections IWL and IWE are more rigorous than those performed pursuant to Regulatory Guide 1.163 and NEI 94-01. For example, Subarticle IWE-3510.1 requires the general visual examination to be the responsibility of an individual who is knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. The subsection also requires the examination to be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation.

Similarly, Subarticle IWL-2320 states that:

> "The Responsible Engineer shall be a Registered Professional Engineer experienced in evaluating the inservice condition of structural concrete. The Responsible Engineer shall have knowledge of the design and Construction Codes and other criteria use in design and construction of concrete containments in nuclear power plants.
>
> The Responsible Engineer shall be responsible for the following:
> (a) development of plans and procedures for examination of concrete surfaces;
> (b) approval, instruction, and training of concrete examination personnel
> (c) evaluation of examination results;
> (d) preparation of repair procedures;
> (e) submittal of report to the Owner documenting results of examinations and repairs."

Based on the above, the Responsible Engineer will ensure that a comprehensive visual examination of the concrete is performed in accordance with Code requirements except where relief has been granted by the NRC. Furthermore, with respect to examinations performed pursuant to both Subsections IWL and IWE, visual examinations of both the concrete surfaces and the liner plate must be reviewed by an Inspector employed by a State or municipality of the United States or an Inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance, in accordance with IWA-2110 and IWA-2120. The combination of the Code requirements for the rigor of the visual examinations plus the third-party review will more than offset the fact one fewer visual examination of the concrete will be performed during a 10-year interval. The fact that the concrete visual examination pursuant to Subsection IWL may be performed during power operation as opposed to during a refueling outage will have no effect on the quality of the examination and will provide flexibility in scheduling of the visual examinations.

On October 6, 2000, the NRC approved relief requests (RR-L1 through RR-L4) associated with containment inservice inspection requirements for PVNGS.

Specifically, APS requested relief from the requirements specified in Subsection IWL of Section XI of the ASME Code. The approved relief requests included relief from the examination schedules in the Code (RR-L3 and RR-L4).

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

APS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed 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 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 required by Regulatory Guide 1.163, "Performance- Based Containment Leak-Testing Program," 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. 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 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. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

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 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.

Conclusion:

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

5.2 Applicable Regulatory Requirements/Criteria

The regulatory basis for TS 3.6.1, "Containment", is to ensure that the primary containment is capable of withstanding the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate. This ensures that offsite radiation exposures are maintained within the limits of 10CFR100.

10 CFR 50, Appendix A, General Design Criterion 16, "Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

This Technical Specification change will not reduce the leak-tightness of the containment. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

APS has evaluated the proposed amendment and determined the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed amendment is not required.

7.0 PRECEDENT

The NRC issued License Amendments 310, 310, and 310 for Oconee Nuclear Station, Units 1, 2 and 3 on January 18, 2000, License Amendments 240 and 214 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 on January 30, 2001, and License Amendments 122 and 100 for Vogtle Electric Generating Plant, Units 1 and 2 on June 6, 2001. The Amendments for Oconee and Calvert Cliffs changed the Concrete Containment Tendon Surveillance Program from being in accordance with Regulatory Guide 1.35 to ASME Code Section XI, Subsection IWL. Additionally, the provisions of SR 3.0.2 were deleted. The Amendments for Vogtle added an exception to the Containment Leakage Rate Testing Program for Regulatory Guide 1.1.63. The exception allows the use of ASME Code Section XI, Subsection IWL for the inspection of concrete surfaces.

8.0 REFERENCES

1. 10 CFR 50.55a

2. Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program."

3. Letter dated January 18, 2000, to W. R. McCollum, Jr., Duke Energy Corporation, "Oconee Nuclear Station Units 1, 2, and 3 RE: Issuance of Amendments (TAC Nos. MA6568, MA6569, and MA6570)." Amendment Nos. 310.

4. Letter dated June 6, 2001, to J. B. Beasley, Jr., Southern Nuclear Operating Company, Inc, "Vogtle Electric Generating Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB1097 and MB1098)." Amendment Nos. 122 and 100.

5. Letter dated January 30, 2001, to C. H. Cruse, Constellation Nuclear, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 RE: Containment Tendon Surveillance Program – Amendment (TAC Nos. MB0011 and MB0012)." Amendment Nos. 240 and 214.

Attachment 1

Proposed Technical Specification Changes (mark-up)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary;

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, as described in Section 1.8 of the UFSAR.

INSERT 1

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)

Insert Page
Technical Specification 5.5.6

Insert 1

ASME Code Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

INSERT 1

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig. The containment design pressure is 60 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

(continued)

Insert Page
Technical Specification 5.5.16

Insert 1

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWL, except where relief has been authorized by the NRC. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Attachment 2

Proposed Technical Specification Pages (retyped)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary;

1. For noble gases: less than or equal to a dose rate of 500 mrems/yr to the total body and less than or equal to a dose rate of 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate of 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public beyond the site boundary due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9.1.1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ASME Code Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (continued)

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWL, except where relief has been authorized by the NRC. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Code Section XI, Subsection IWE, except where relief has been authorized by the NRC.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 52.0 psig for Units 1 and 3, and 58.0 psig for Unit 2. The containment design pressure is 60 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1 % of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq$14.5 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Attachment 3

Changes to Technical Specification Bases Pages

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and C leakage and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

(continued)

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Technical Specification Bases 3.6.1
SR 3.6.1.1

Insert 1

The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).

INSERT 1 →

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 3.8.
3. UFSAR, Section 6.2.
4. Regulatory Guide 1.35, Revision 1 and Draft of Revision 3.

INSERT 2 →

Insert Page
Technical Specification Bases 3.6.1
SR 3.6.1.2

Insert 1

in accordance with ASME Code Section XI, Subsection IWL (Ref. 4) and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

Insert 2

ASME Code Section XI, Subsection IWL.