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SUBJECT: Responds to NRC request for addl info re 940218 proposed amend to TSs 5.3.1 & 5.6.1 & adding new TS Section 3/4.9.13 & Bases 3/4.9.13.

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WILLIAM L. STEWART
EXECUTIVE VICE PRESIDENT
NUCLEAR

102-03012-WLS/RAB/GEC
June 20, 1994

U. S. Nuclear Regulatory Commission
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Reference: (1) Letter 102-02838, dated February 18, 1994, from W. F. Conway, Executive Vice President, Nuclear, APS, to NRC, Proposed Amendment to Technical Specification Sections 5.3.1 and 5.6.1, and new Technical Specification Section 3/4.9.13 and BASES 3/4.9.13

(2) Letter, dated April 13, 1994, from L. N. Tran, NRC, to W. F. Conway, Executive Vice President, Nuclear, APS, Spent Fuel Pool Modification - Palo Verde Nuclear Generating Station (TAC Nos. M88992, M88991, and M88993)

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Responses to NRC Request for Additional Information Regarding Proposed Amendment to Technical Specification Sections 5.3.1 and 5.6.1, and New Technical Specification Section 3/4.9.13 and BASES 3/4.9.13
File: 94-056-026; 94-005-419.05

Arizona Public Service Company (APS) submitted a proposed amendment to Technical Specification (TS) Sections 5.3.1, Fuel Assemblies, and 5.6.1, Criticality, and new Technical Specification Section 3/4.9.13, Boron Concentration - Storage Pool, and BASES 3/4.9.13, Boron Concentration - Storage Pool, in Reference 1.

The Enclosure to this letter contains responses to the NRC staff request for additional information (Reference 2). This clarifying information is being provided to the Arizona Radiation Regulatory Agency (ARRA) by copy of this letter.

A001

Should you have any questions, please contact Richard A. Bernier at (602) 393-5882.

Sincerely,

WL Stewart
6/20/94

WLS/RAB/GEC/dd

Enclosure

cc: L. J. Callan
K. E. Perkins
K. E. Johnston
B. E. Holian
A. V. Godwin (ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, W. L. Stewart, represent that I am Executive Vice President - Nuclear, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true and correct.

W L Stewart

W. L. Stewart

Sworn to Before Me This 20th Day of June, 1994.

Ramona Wright

Notary Public

My Commission Expires

OFFICIAL SEAL
RAMONA WRIGHT
Notary Public - State of Arizona
MARICOPA COUNTY
My Commission Expires Aug. 15, 1997

ENCLOSURE

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED AMENDMENT TO

TECHNICAL SPECIFICATION SECTIONS 5.3.1 AND 5.6.1,

AND NEW TECHNICAL SPECIFICATION SECTION 3/4.9.13

AND BASES 3/4.9.13

Question 1:

Provide the name of the organization that performed the criticality calculations.

Response:

ABB Combustion Engineering (ABB-CE)

Question 2:

What organization performed the qualification of the analytical methods.

Response:

ABB-CE

Question 3:

In view of the previous discrepancies discovered in the analysis of the Millstone 2 spent fuel pool, discuss the acceptability of the use of these methods (CEPAK code) for the criticality analysis for Palo Verde.

Response:

The discrepancies discovered in the criticality analysis of the Millstone 2 spent fuel pool were:

A. The use of unshielded epithermal B-10 cross sections to represent a Boroflex poison box.

B. The use of geometric bucklings in CEPAK which were not indicative of the neutronic environment produced by the highly poisoned Boroflex fuel rack.

The Palo Verde Nuclear Generating Station (PVNGS) spent fuel racks do not employ any type of boron material for criticality safety. In addition, ABB-CE has revised the CEPAK-DOT methodology for all spent fuel rack calculations and demonstrated that the methodology has a relatively low bias (0.00197 delta K-effective units) and 95/95 calculational uncertainty (0.00714 delta K-effective units) by comparison to a set of 41 critical experiments.

Question 4:

In Section 3.1 of your submittal, you referenced Table 5-1 that contains the Assembly Burnup-Initial Enrichment data; please provide this table.

Response:

The information in "Table 5-1" was inadvertently omitted from the amendment submittal. This information, which is part of Section 3.1 and was intended to be included within the text as "Table 1," is provided below and should be considered part of the original submittal:

TABLE 1

REQUIRED ASSEMBLY BURNUP FOR STORAGE IN REGION 2 AND REGION 3

INITIAL ENRICHMENT (weight percent)	REGION 2 ASSEMBLY BURNUP (Mwd/mtu)	REGION 3 ASSEMBLY BURNUP (Mwd/mtu)
2.5	7,302	19,846
3.5	18,787	32,241
4.5	28,776	43,797

Question 5:

Provide a discussion on how the uncertainty in the depletion calculations was determined and included in the analysis.

Response:

A constant uncertainty equal to 0.0081 delta K-effective units was included in all of the spent fuel rack calculations to account for the uncertainty in assembly burnup. The uncertainty associated with assembly burnup was included in the square root of the sum of uncertainties squared. This reactivity uncertainty was determined by multiplying a constant burnup uncertainty of 1,000 megawatt days/metric ton (Mwd/mtu) by a calculated ratio of delta spent fuel reactivity and delta assembly burnup. The 1,000 Mwd/mtu uncertainty in assembly burnup is a conservative estimate of the 95/95 uncertainty in assembly burnup.

In addition to the uncertainty in the integrated assembly burnup, an additive uncertainty equal to 0.007 delta K-effective units was included in all of the calculational results to account for non-uniform axial burnup distributions. The uncertainty due to non-uniform axial burnup distributions was added directly to the DOT calculated multiplication factors.

Question 6:

On page 7 of 9 of the submittal, please clarify the minimum monolith thickness uncertainty.

Response:

The monolith thickness uncertainty is +/- 0.006 inches.

Question 7:

Provide the range of temperature variation that was used in the uncertainty analysis.

Response:

The nominal moderator temperature employed in the criticality calculations is 98.6°F. The moderator temperatures employed for the uncertainty analysis ranged from 68°F up to 248°F.

Question 8:

The proposed new wording of TS 5.6.1.1.b should be between "adjacent storage <u>cell</u> locations" rather than "adjacent storage <u>rack</u> locations." If you agree, provide a revised TS page in your response.

Response:

A replacement page indicating "adjacent storage <u>cell</u> locations" rather than "adjacent storage <u>rack</u> locations" is provided for each unit as an attachment to this enclosure. This change in the wording of proposed new TS 5.6.1.1.b is not considered a substantive change and is viewed to be on the order of an editorial modification.

Question 9:

Since TS 5.6.1.2 was only applicable for the first core loading, provide a discussion as to why it has not been replaced by a TS representative of the new (unirradiated) dry storage racks, which have been analyzed for up to 4.30 weight percent U-235 assemblies.

Response:

TS 5.6.1.2 reflects dry storage of new fuel in the spent fuel storage pool prior to filling of the spent fuel storage pool (i.e., before introduction of irradiated fuel into the pool). Arizona Public Service Company (APS) agrees that this requirement was applicable only to the first core loading in each unit and with the consequent deletion of TS 5.6.1.2. No TS representative of the new (unirradiated) dry storage racks is planned as the new fuel storage racks are not currently included in the PVNGS TS.

Question 10:

Since the criticality analysis was performed only for initial assembly enrichments up to 4.30 weight percent, TS Figure 5.6-1 should only extend to 4.30 weight percent instead of the 5.0 weight percent upper limit shown.

Response:

A revised TS Figure 5.6-1 showing the boundary curves between the regions extending over the range of initial enrichments from 2.00 to 4.30 weight percent is provided for each unit as an attachment to this enclosure. This change in TS Figure 5.6-1 is not considered a substantive change as those portions of the curves shown on the revised Figure 5.6-1 have not changed from the original submittal.

Question 11:

Since there are many individual fuel rod distributions that could give a radially averaged enrichment of 4.30 weight percent U-235 but that could result in different storage rack reactivities, discuss why your calculations are considered to be bounding.

Response:

The 16X16 fuel assembly contains 5 large water holes. The fuel cells adjacent to the water holes experience a more thermalized spectrum of neutrons compared to the assembly averaged spectrum. In addition, but to a lesser degree, the fuel pins in the four corners of the fuel assembly experience a more thermalized neutron spectrum. In order to flatten the intra-assembly power distribution during operation, two different fuel enrichments are employed within the fuel assembly. The enrichment

of the fuel pins surrounding the water holes and in the corners of the fuel assembly is lower than the enrichment of the remaining fuel pins.

Calculations were performed to investigate the reactivity effect of a uniform enrichment distribution versus the described zoned fuel enrichment distribution. The zoned fuel enrichment distribution considered zoning around the water holes and in the corners of the fuel assembly. The results of these calculations demonstrate that the uniform distribution of enrichment produces a more reactive fuel assembly. Since all of the criticality analysis calculations were performed with uniform fuel enrichments, the results bound the case with the above described zoned fuel enrichments.