10 CFR 50.90

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102-08142-BJR/MDD
August 21, 2020

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

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station Units 1, 2, and 3**
Docket Nos. STN 50-528, 50-529, and 50-530
Renewed Operating License Number NPF-41, NPF-51, and NPF-74
Application to Revise Technical Specifications to Make Necessary Administrative Changes

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) is submitting a request for an amendment to the Technical Specifications (TS) for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3.

APS requests five (5) administrative changes to the TS that remove no longer applicable information, extraneous information and adopt standard industry terminology. They include:

1. Surveillance Requirement (SR) 3.1.5.3 – Removal of a one-time use note for Unit 2 control element assembly (CEA) number 88
2. TS 3.7.17, 4.3, and 5.5.21 – Removal of no longer applicable pages related to the implementation of Amendment 203, which addressed a revised spent fuel pool (SFP) criticality analysis
3. TS 4.1 – Removal of extraneous information from the *Site Location* description
4. TS 5.5.2 – Removal of remaining post-accident sampling subsystem (PASS) reference
5. TS 5.7 – Modification of radiation protection terminology to match industry standards

The enclosure provides a description and assessment of the proposed changes. Attachment 1 to the enclosure provides the existing TS pages marked up to show the proposed changes. Attachment 2 to the enclosure provides revised (clean) TS pages. Attachment 3 to the enclosure provides marked up TS Bases pages to show the proposed changes. The changes to the TS Bases are provided for information only.

A pre-submittal meeting for these changes was held between APS and the NRC staff on August 12, 2020. Approval of the proposed amendment is requested by August 21, 2021. Once approved, the amendment will be implemented within 90 days.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board has reviewed and approved this license amendment request (LAR). By copy of this letter, the LAR is being forwarded to the Arizona Department of Health Services – Bureau of Radiation Control in accordance with 10 CFR 50.91(b)(1).

No new commitments are being made to the NRC by this letter.

Should you need further information regarding this letter, please contact Matthew S. Cox, Licensing Section Leader, at (623) 393-5753.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: August 21, 2020
(Date)

Sincerely,

Rash, Bruce (Z77439)

Digitally signed by Rash, Bruce (Z77439)
DN: cn=Rash, Bruce (Z77439)
Date: 2020.08.21 13:37:00 -07'00'

BJR/MDD/mg

Enclosure: Description and Assessment of Proposed License Amendment

cc:	S. A. Morris	NRC Region IV Regional Administrator
	S. P. Lingam	NRC NRR Project Manager for PVNGS
	C. A. Peabody	NRC Senior Resident Inspector for PVNGS
	B. Goretzki	Arizona Department of Health Services – Bureau of Radiation Control

ENCLOSURE

Description and Assessment of Proposed License Amendment

Enclosure
Description and Assessment of Proposed License Amendment

Subject: License Amendment Request to Make Miscellaneous Administrative Changes

Table of Contents

LIST OF ACRONYMS

APS	Arizona Public Service Company
CE	Combustion Engineering
CEDM	Control Element Drive Mechanism
CEA	Control Element Assembly
CFR	Code of Federal Regulations
LA	License Amendment
LAR	License Amendment Request
LCO	Limiting Condition for Operation
LDCR	License Document Change Request
NISP	Nuclear Industry Standard Process
PASS	Post-Accident Sampling Subsystem
PVNGS	Palo Verde Nuclear Generating Station
REP	Radiation Exposure Permit
RP	Radiation Protection
RWP	Radiation Work Permit
SFP	Spent Fuel Pool
SR	Surveillance Requirement
TS	Technical Specifications
TSTF	Technical Specifications Task Force
UFSAR	Updated Final Safety Analysis Report
UGC	Upper Gripper Coil

1.0 SUMMARY DESCRIPTION

Arizona Public Service Company (APS) is requesting a license amendment (LA) to the Palo Verde Nuclear Generating Station Units 1, 2 and 3 Technical Specifications (TS). The proposed amendment would modify the TS by making various administrative changes.

This includes:

1. Surveillance Requirement (SR) 3.1.5.3 – Removal of a one-time use note for Unit 2 control element assembly (CEA) number 88
2. TS 3.7.17, 4.3, and 5.5.21 – Removal of no longer applicable pages related to the implementation of Amendment 203, which addressed a revised spent fuel pool (SFP) criticality analysis
3. TS 4.1 – Removal of extraneous information from the *Site Location* description
4. TS 5.5.2 – Removal of remaining post-accident sampling subsystem (PASS) reference
5. TS 5.7 – Modification of radiation protection terminology to match industry standards

2.0 DETAILED DESCRIPTION

2.1 Unit 2 CEA Number 88 SR Note Removal (SR 3.1.5.3)

2.1.1 Description of the Proposed Change

This license amendment request (LAR) proposes to revise TS Surveillance Requirement (SR) 3.1.5.3 to remove a one-time historical note, which was modified by exigent LA 196 (Reference 6.1) on September 25, 2015. The note was only applicable to Unit 2 CEA 88 and for a portion of one operating cycle.

The note states:

> *Not required to be performed for Unit 2 CEA 88 for the remainder of Cycle 19.*

2.1.2 Reason for the Proposed Change

The one time note, which excluded Unit 2 CEA 88 from the requirements of SR 3.1.5.3, is no longer necessary. During the Unit 2 Fall of 2015 refueling outage (2R19), the degraded control element drive mechanism (CEDM) upper gripper coil (UGC) that holds CEA 88 was replaced. Since these repairs were made, Unit 2 CEA 88 has been tested and found in compliance with SR 3.1.5.3. This proposed LAR will restore PVNGS Technical Specification SR 3.1.5.3 to its original content and applicability.

Therefore, the note is no longer applicable and APS requests

removal of this note.

2.2 SFP Criticality Analysis and Physical Changes (TS 3.7.17, 4.3, and 5.5.21)

2.2.1 Description of the Proposed Change

This proposed LAR will modify PVNGS TS Sections 3.7.17, *Spent Fuel Assembly Storage*, 4.3, *Fuel Storage*, and 5.5.21, *Spent Fuel Storage Rack Neutron Absorber Monitoring Program*, to remove the "Before SFP transition" pages and revise the "After SFP transition" pages.

2.2.2 Reason for the Proposed Change

The PVNGS spent fuel pool (SFP) was modified over a two (2) year period in accordance with LA 203 (Reference 6.2) to incorporate the results of an updated criticality safety analysis for both new and spent fuel storage and to install the NETCO-SNAP-IN® neutron absorbing rack inserts into select spent fuel pool storage rack cells. The transition to the new SFP configuration was completed in all three units in accordance with the SFP Transition Plan on December 31, 2019. With all three PVNGS unit SFPs fully transitioned, this proposed LAR will remove the no longer applicable "Before SFP transition" pages and revise the "After SFP transition" pages. This LAR will revise PVNGS TS to now only contain guidance related to the new SFP criticality analysis and configurations.

2.3 Site Location Description Administrative Changes (TS 4.1)

2.3.1 Description of the Proposed Change

This proposed LAR will modify PVNGS TS Section 4.1, *Site Location*, to remove erroneous and extraneous information. This proposed LAR requests removal of the actual acreage number, site elevation numbers and the minimum distance from the nearest containment building to the exclusion area boundary from TS Section 4.1.

2.3.2 Reason for the Proposed Change

The NRC Standard Technical Specifications for Combustion Engineering (CE) Plants (NUREG-1432, Volume 1, Revision 4) does not contain specific guidance for what information is to be provided in TS Section 4.1, other than the section heading "Site Location" and the following bracketed information:

[Text description of site location.]

10 CFR 50.36, "Technical Specifications," establishes the regulatory requirements related to the content of TS. Paragraph 50.36(a)(1) requires an application for an operating license to

include proposed TS. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the TS. Pursuant to 10 CFR 50.36, TS for operating reactors are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. In accordance with 10 CFR 50.36(c)(4), Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories (1), (2), and (3) discussed above.

The site specific acreage, site elevation details, and exclusion area boundary information is not relevant to 10 CFR 50.36(c)(4), *Design features*, as this information does not have a significant effect on safety which is the criterion for inclusion in TS Section 4. The PVNGS Updated Final Safety Analysis Report (UFSAR) Sections 2.1.1.2, *Site Area Maps,* and 2.1.1.3, *Boundaries for Establishing Effluent Release Limits*, contain the site specific acreage, site elevation details, and exclusion area boundary information. Additionally, the exclusion area boundary is discussed in the PVNGS Emergency Plan.

2.4 Final Removal of PASS Components Reference (TS 5.5.2)

2.4.1 Description of the Proposed Change

This proposed LAR is the final administrative TS change to remove discussion of the elements of post-accident sampling subsystem (PASS) from TS Section 5.5.2, *Primary Coolant Sources Outside Containment*. This section includes reference to the PASS reactor coolant and containment atmosphere sampling portions. Additionally, a note adds, "until such time as a modification eliminates the PASS penetration as a potential leakage path." This proposed LAR requests the removal of both the reference to the PASS equipment and the associated note in TS 5.5.2.

2.4.2 Reason for the Proposed Change

The PASS system was approved for abandonment in LA 136 (reference 6.3) dated September 28, 2001. The LA was structured to permit removal of the PASS equipment over time. PVNGS has since removed, capped, and/or abandoned in place any remaining physical connections of the PASS piping, associated valves and pipe hangers.

The PASS removal or abandonment efforts are complete and referenced PASS components in TS Section 5.5.2 are no longer

applicable and should be removed. Therefore, APS requests the removal of PASS references in TS Section 5.5.2.

2.5 Radiation Protection Terminology and Position Title Changes (TS 5.7)

2.5.1 Description of the Proposed Change

This proposed LAR requests that Radiation Exposure Permit (REP) be replaced with the PVNGS site and industry standards terminology Radiation Work Permit (RWP). Additionally, APS requests that any reference to radiation protection (RP) specific positions be removed and replaced with the generic term of "radiation protection supervision."

2.5.2 Reason for the Proposed Change

PVNGS has aligned plant radiation protection documents to use the common industry accepted terminology of RWP discussed in Nuclear Industry Standard Process (NISP) document, NISP-RP-013 (Reference 6.4), and a more generic positional naming convention for radiation protection supervision discussed in Technical Specification Task Force (TSTF)-65 (Reference 6.5). These changes align PVNGS to common industry terminology, allow necessary operational flexibility and minimize future administrative TS revisions.

3.0 TECHNICAL EVALUATION

3.1 Unit 2 CEA Number 88 SR Note Removal (SR 3.1.5.3)

The time limit of the one-time license amendment has expired and the note is no longer applicable. This change will restore PVNGS Technical Specification SR 3.1.5.3 to its original content and applicability. Removal of a one-time use note for Unit 2 control element assembly number 88 is an administrative change.

3.2 SFP Criticality Analysis and Physical Changes (TS 3.7.17, 4.3 and 5.5.21)

Removal of the "Before SFP transition" pages is administrative. These pages were only necessary during the transitional phase of the SFP physical changes. With all SFP configuration changes complete, the "Before SFP transition" pages are no longer applicable or necessary.

3.3 Site Location Description Administrative Changes (TS 4.1)

Modifying TS 4.1, *Site Location*, content is administrative. Removal of extraneous information, which is already available in PVNGS UFSAR and PVNGS Emergency Plan, will minimize unnecessary future TS changes. The *Site Location* in TS Section 4.1 is descriptive information that establishes a general context for the site with regard to the large population center and is not relevant to 10 CFR 50.36(c)(4), *Design features*, as this information does not have a significant effect on

safety, which is the criterion for inclusion in TS Section 4. The PVNGS TS Sections 4.2 and 4.3 deal with geometric arrangements (e.g., fuel types in the core and fuel storage design features) that have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of 10 CFR 50.36 (i.e., *Safety limits, Limiting conditions for operation and Surveillance requirements*).

PVNGS maintains this information in the UFSAR and the PVNGS Emergency Plan which are controlled pursuant to 10 CFR 50.71(e), 10 CFR 50.59 and 10 CFR 50.54(q).

3.4 Final Removal of PASS Components Reference (TS 5.5.2)

Removal of reference to PASS components previously removed or abandoned in place is administrative. Previous LA removed PASS from the PVNGS license basis. Removing these PASS references from the TS will eliminate a potential source of confusion to plant operators.

3.5 Radiation Protection Terminology and Position Title Changes (TS 5.7)

Proposed changes to TS 5.7, *High Radiation Area*, are administrative. PVNGS has adopted the use of the standard industry terminology of RWP and has aligned related plant documents accordingly. Changing the terminology does not affect the control of high radiation areas. The proposed change also transitions PVNGS to a more generic radiation protection personnel listing. These changes are administrative and do not change any technical aspects of the facility.

4.0 REGULATORY EVALUATION

4.1 Precedent

A recent precedent for a request to make miscellaneous administrative changes is Watts Bar Nuclear Plant, Units 1 and 2. NRC issued LA 135 and 39 dated June 22, 2020 (Reference 6.6), to revise several TS changes. These changes included removal of one-time applicable LAs and administrative changes associated with site location. The other administrative changes are unique to Watts Bar Nuclear Plant, Units 1 and 2.

4.2 Significant Hazards Consideration

Arizona Public Service Company (APS) is requesting an amendment to Facility Operating Licenses NPF-41, NPF-51, and NPF-74 for Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2 and 3, respectively. The proposed amendment would modify Technical Specifications (TS) by making various administrative changes. This includes:

1. Surveillance Requirement (SR) 3.1.5.3 – Removal of a one-time use note for Unit 2 control element assembly (CEA) number 88

2. TS 3.7.17, 4.3, and 5.5.21 – Removal of no longer applicable pages related to the implementation of Amendment 203, which addressed a revised spent fuel pool (SFP) criticality
3. TS 4.1 – Removal of extraneous information from the *Site Location* description
4. TS 5.5.2 – Removal of remaining post-accident sampling subsystem (PASS) reference
5. TS 5.7 – Modification of radiation protection terminology to match industry standards

These administrative changes can be grouped into three (3) categories to determine the proposed changes do not involve a significant hazards consideration. These categories include:

- Removal of no longer applicable material
- Removal of extraneous information
- Transition to industry accepted terminology

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

   Response: No.

   The proposed changes are administrative. The removal of no longer applicable notes or descriptions and extraneous information and transitioning to industry-accepted terminology aligns the TS with current plant configurations, provides clarity to avoid operator misinterpretation, and allows consistency across PVNGS documentation and procedures. Removal of no longer applicable exceptions and no longer applicable descriptions restore the TS to full applicability and current plant configurations.

   The administrative changes return TS descriptions to align with current plant configurations approved in previous LAs and restore original surveillance requirements to their original full applicability. Therefore, these changes do not involve a significant increase in the probability or consequence 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 are administrative. The removal of no longer applicable notes or descriptions and extraneous information and transitioning to industry accepted terminology aligns TS with current plant configurations, provides clarity to avoid operator

misinterpretation, and allows consistency across PVNGS documentation and procedures. Removal of no longer applicable exceptions and no longer applicable descriptions restore the TS to full applicability and current plant configurations.

The administrative changes return TS descriptions to align with current plant configurations approved in previous LAs and restore original surveillance requirements to their original full applicability. 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. Therefore, these changes do not involve a significant increase in the probability or consequence of an accident previously evaluated.

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

   Response: No.

   The proposed changes are administrative. The removal of no longer applicable notes or descriptions and extraneous information and transitioning to industry-accepted terminology aligns the TS with current plant configurations, provides clarity to avoid operator misinterpretation, and allows consistency across PVNGS documentation and procedures. Removal of no longer applicable exceptions and no longer applicable descriptions restore the TS to full applicability and current plant configurations.

   The administrative changes return TS descriptions to align with current plant configurations approved in previous LAs and restore original surveillance requirements to their original full applicability. The changes are administrative, and will not diminish any organizational or administrative controls currently in place. 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. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, APS concludes that the proposed change 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.

4.3 Conclusion

In conclusion, 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) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL EVALUATION

The proposed changes are administrative and do not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, *Standards for Protection Against Radiation*, or would not change an inspection or surveillance requirement. The proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

6.0 REFERENCES

6.1 NRC issued Amendment No. 196 to Renewed Facility Operating License No. NPF-51 for the Palo Verde Nuclear Generating Station, Unit 2, dated September 25, 2015 [NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML15266A005]

6.2 NRC issued Amendment No. 203 to Renewed Facility Operating Licenses No. NPF-41, NPF-51 and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, dated July 28, 2017 (ADAMS Accession No. ML17188A412)

6.3 NRC issued Amendment No. 136 to Facility Operating Licenses No. NPF-41, NPF-51 and NPF-74 for the Palo Verde Nuclear Generating Station, Units 1, 2, and 3, dated September 28, 2001 (ADAMS Accession No. ML012710441)

6.4 Nuclear Industry Standard Process - Radiological Protection, NISP-RP-013, Radiation Protection Standard Glossary of Terms, Revision 1, dated December 11, 2018

6.5 Technical Specification Task Force (TSTF)–65, Use of Generic Titles for Utility Positions, Revision 1, dated November 10, 1994 (ADAMS Accession No. ML040080572)

6.6 NRC issued Amendment Nos. 135 and 39 for the Watts Bar Nuclear Plant Units 1 and 2 regarding Miscellaneous Administrative Changes to the Technical Specifications, dated June 22, 2020 (ADAMS Accession No. ML20016A278)

ATTACHMENT 1:

Proposed Technical Specification Changes (Mark-Up)

Changed Page(s)

3.1.5-3
3.7.17-1
3.7.17-1a
3.7.17-2
3.7.17-2a
3.7.17-3
3.7.17-3a
3.7.17-4
3.7.17-4a
3.7.17-5a
3.7.17-6a
3.7.17-7a
4.0-1
4.0-2
4.0-2a
4.0-3
4.0-3a
5.5-2
5.5-22
5.7-1
5.7-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each full strength and part strength CEA is within 6.6 inches of all other CEAs in its group.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.2	Verify that, for each CEA, its OPERABLE CEA position indicator channels indicate within 5.2 inches of each other.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.3	~~NOTE~~ ~~Not required to be performed for Unit 2 CEA 88 for the remainder of Cycle 19.~~ Verify full strength CEA freedom of movement (trippability) by moving each individual full strength CEA that is not fully inserted in the core at least 5 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.4	Perform a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.5	Verify each full strength CEA drop time $\leq$ 4.0 seconds.	Prior to reactor criticality, after each removal of the reactor head

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3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment, burnup, and decay time of each fuel assembly stored in each of the four regions of the fuel storage pool shall be within the acceptable burnup domain for each region as shown in Figures 3.7.17-1, 3.7.17-2, or 3.7.17-3, and described in Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ----------------------------- Initiate action to move the noncomplying fuel assembly into an appropriate region.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Figures 3.7.17-1, 3.7.17-2, or 3.7.17-3, and Specification 4.3.1.1.	Prior to storing the fuel assembly in the fuel storage pool.

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment, burnup, and decay time of each fuel assembly shall be in compliance with the requirements specified in Tables 3.7.17-1 through 3.7.17-5.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE---------- LCO 3.0.3 is not applicable. ---------------------------- Initiate action to move the noncomplying fuel assembly into an appropriate region.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Tables 3.7.17-1 through 3.7.17-5, Figure 3.7.17-1, and Specification 4.3.1.1.	Prior to storing the fuel assembly in the fuel storage pool.

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Figure 3.7.17-1
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 2

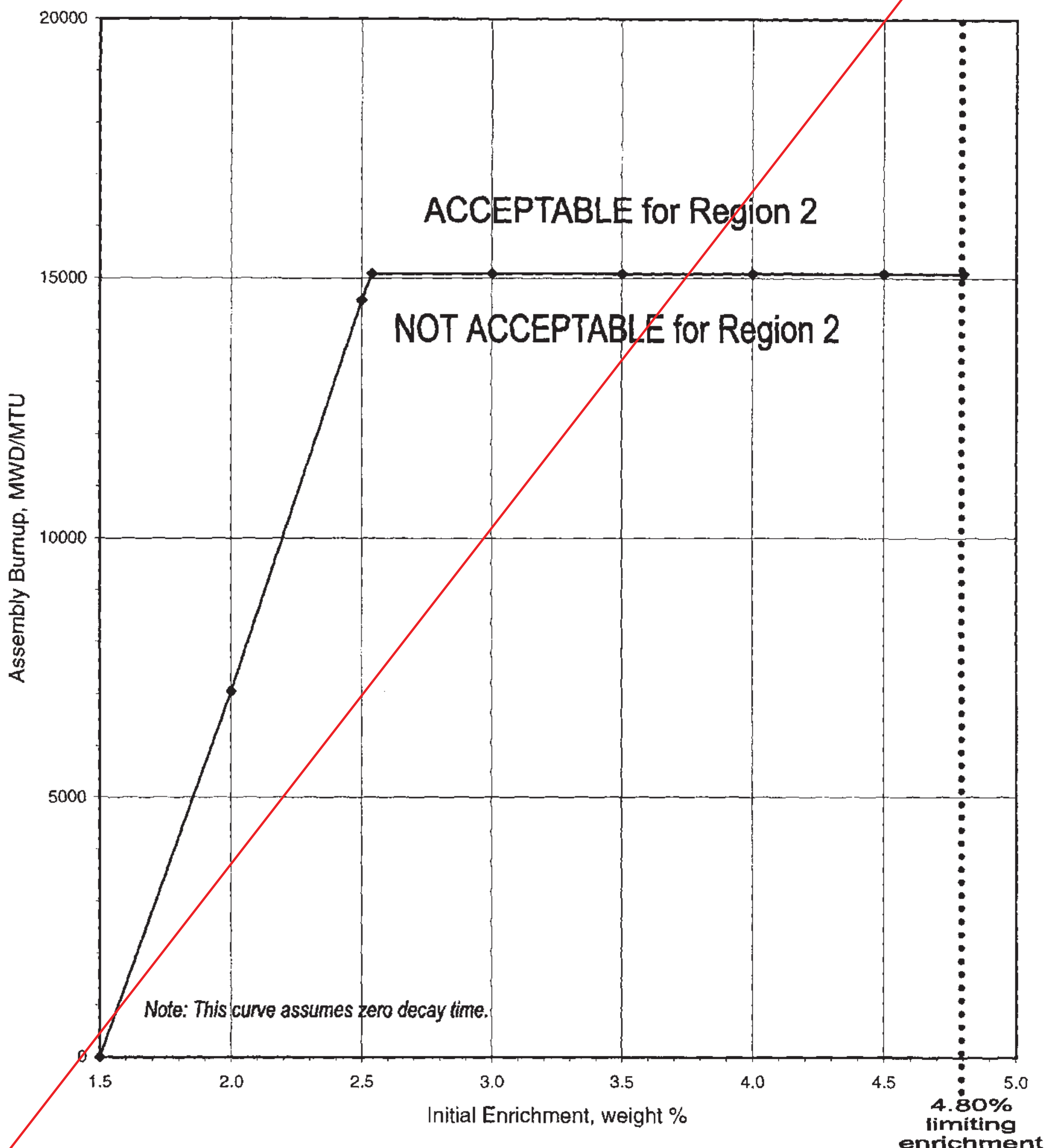

Table 3.7.17-1

Fuel Regions Ranked by Reactivity	
Fuel Region 1	Highest Reactivity (See Note 2)
Fuel Region 2	
Fuel Region 3	
Fuel Region 4	
Fuel Region 5	
Fuel Region 6	Lowest Reactivity

Notes:

1. Fuel Regions are defined by assembly average burnup, initial enrichment[1] and decay time as provided by Table 3.7.17-2 through Table 3.7.17-5.
2. Fuel Regions are ranked in order of decreasing reactivity, e.g., Fuel Region 2 is less reactive than Fuel Region 1, etc.
3. Fuel Region 1 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U. No burnup is required.
4. Fuel Region 2 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U with at least 16.0 GWd/MTU of burnup.
5. Fuel Regions 3 through 6 are determined from the minimum burnup (BU) equation and coefficients provided in Tables 3.7.17-2 through 3.7.17-5.
6. Assembly storage is controlled through the storage arrays defined in Figure 3.7.17-1.
7. Each storage cell in an array can only be populated with assemblies of the Fuel Region defined in the array definition or a lower reactivity Fuel Region.

[1]Initial Enrichment is the nominal ^{235}U enrichment of the central zone region of fuel, excluding axial blankets, prior to reduction in ^{235}U content due to fuel depletion. If the fuel assembly contains axial regions of different ^{235}U enrichment values, such as axial blankets, the maximum initial enrichment value is to be utilized.

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Figure 3.7.17-2
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 3
(at decay times from 0 to 20 years)

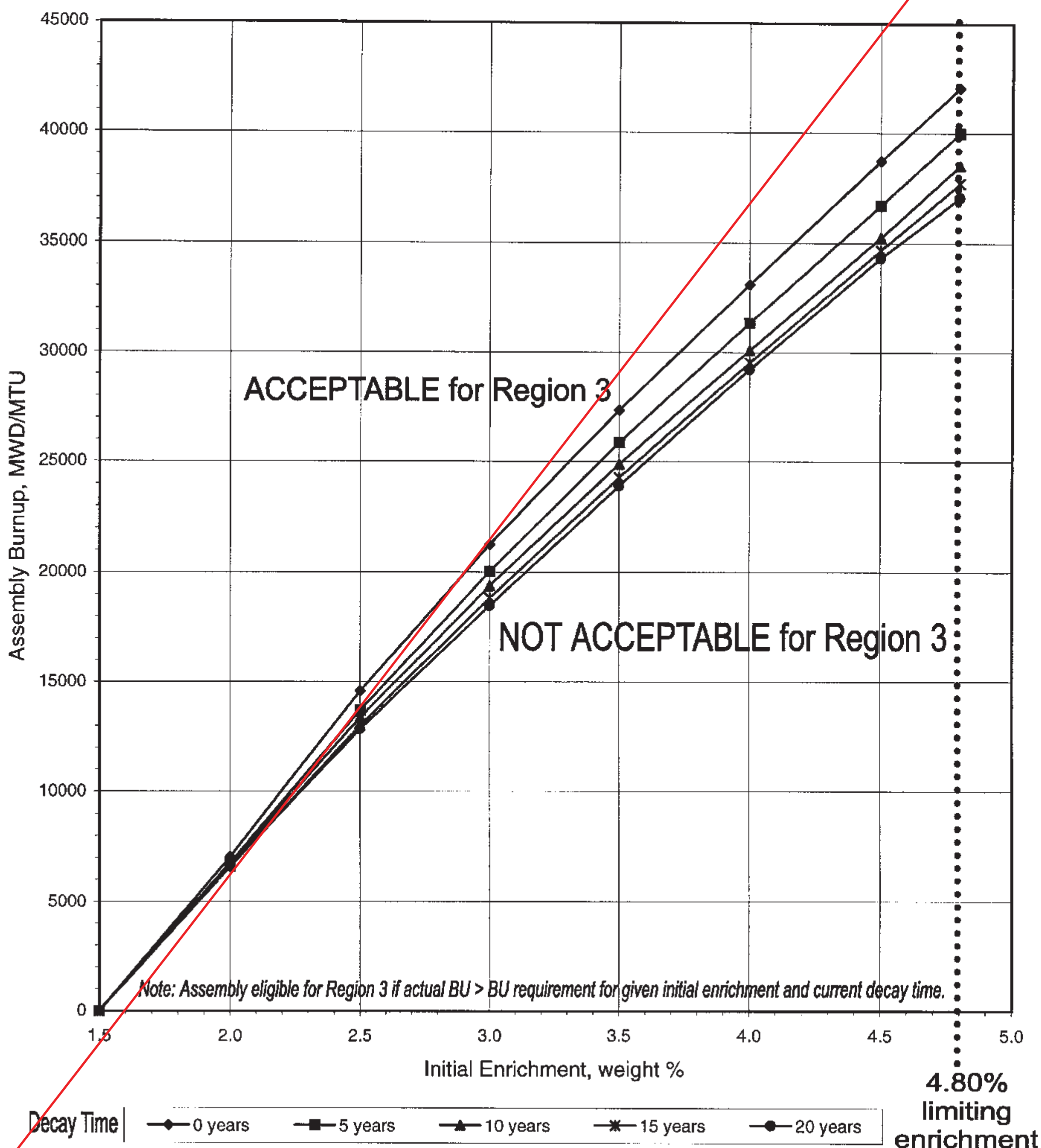

Table 3.7.17-2

Fuel Region 3: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	-0.8100	6.5551	-2.9050	-21.0499
5	-0.9373	7.6381	-6.0246	-18.0299
10	-0.8706	6.8181	-3.1913	-21.0299
15	-0.7646	5.6311	0.7657	-25.1599
20	-0.7233	5.1651	2.3084	-26.7499

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 2.50 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 2.50 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

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Figure 3.7.17-3
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 4
(at decay times from 0 to 20 years)

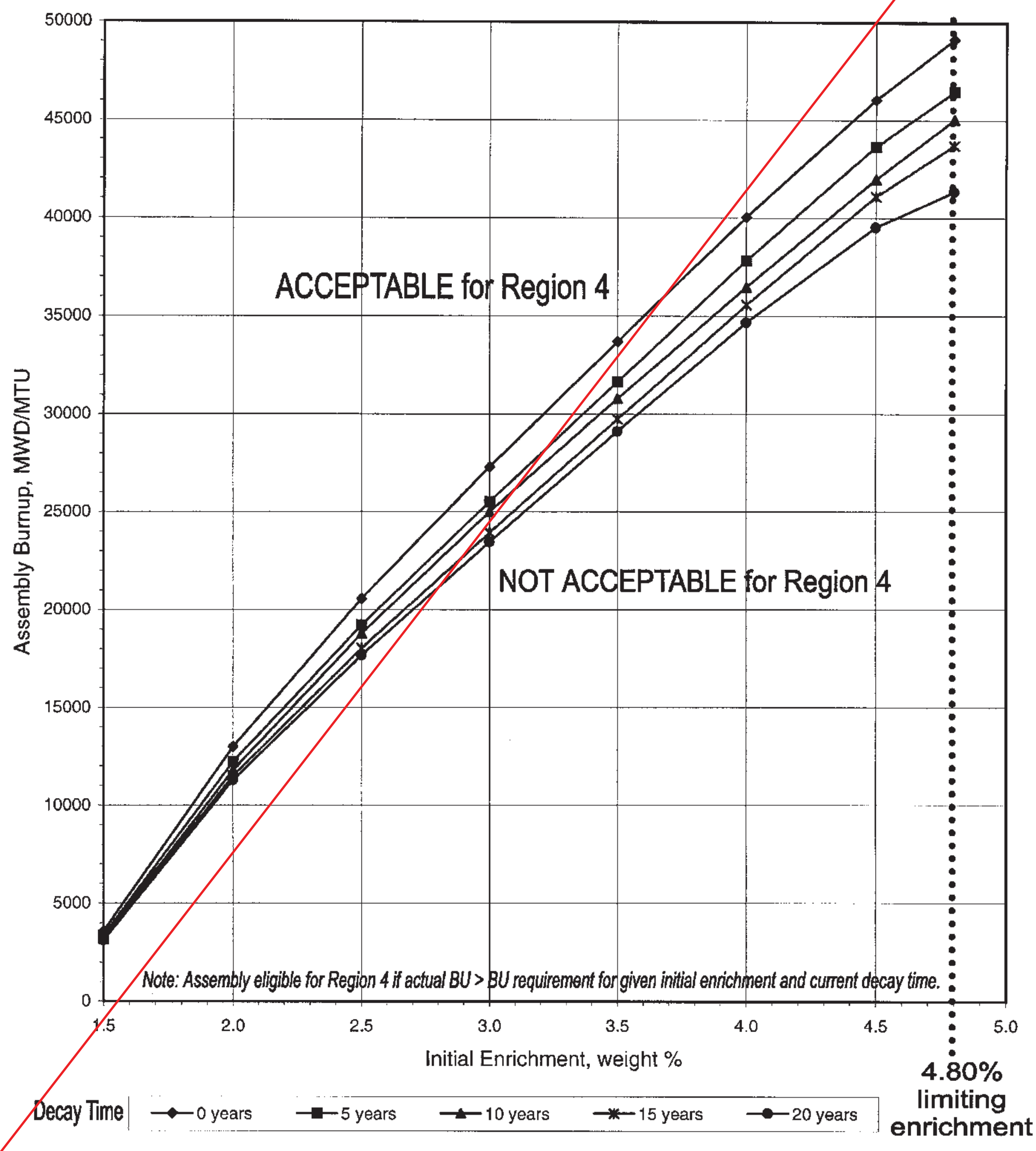

Table 3.7.17-3

Fuel Region 4: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.0333	-2.1141	27.4985	-41.8258
5	-0.2105	0.2472	19.7919	-34.2641
10	0.0542	-2.5298	28.0953	-41.7092
15	0.3010	-5.0718	35.6966	-48.5494
20	0.4829	-6.9436	41.3118	-53.6182

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.75 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.75 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Table 3.7.17-4

Fuel Region 5: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.1586	-3.0177	28.7074	-39.8636
5	-0.2756	1.3433	14.5578	-26.4388
10	-0.2897	1.3218	14.6176	-26.4160
15	-0.0736	-0.9107	21.2118	-32.1887
20	0.1078	-2.7684	26.6911	-36.9873

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.65 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.65 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Table 3.7.17-5

Fuel Region 6: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.4890	-6.7447	42.7619	-49.3143
5	0.5360	-6.9115	41.1003	-46.6977
10	0.4779	-6.1841	37.6389	-43.0309
15	0.4575	-5.8844	35.8656	-41.0274
20	0.3426	-4.7050	31.8126	-37.2800

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.45 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.45 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Figure 3.7.17-1
Allowable Storage Arrays

Array A Two Region 1 assemblies (1) checkerboarded with two blocked cells (X). The Region 1 assemblies are each in a cell with a stainless steel L-insert. No NETCO-SNAP-IN® inserts are credited.	**1**	**X**
	X	**1**
Array B Two Region 1 assemblies (1) checkerboarded with two cells containing trash cans (TC). The Region 1 assemblies are each in a cell with a stainless steel L-insert. Every cell without a stainless steel L-insert must contain a NETCO-SNAP-IN® insert.	**1**	**TC**
	TC	**1**
Array C Two Region 2 assemblies (2) checkerboarded with one Region 3 assembly (3) and one blocked cell (X). The Region 2 assemblies are each in a cell with a stainless steel L-insert. The Region 3 assembly is in a cell containing a NETCO-SNAP-IN® insert.	**2**	**X**
	3	**2**
Array D One Region 2 assembly (2) checkerboarded with three Region 4 assemblies (4). The Region 2 assembly and the diagonally located Region 4 assembly are each in a storage cell with a stainless steel L-insert. The two storage cells without a stainless steel L-insert contain a NETCO-SNAP-IN® insert.	**2**	**4**
	4	**4**
Array E Four Region 5 assemblies (5). Two storage cells contain a stainless steel L-insert. One cell contains a NETCO-SNAP-IN® insert. One storage cell contains no insert.	**5**	**5**
	5	**5**
Array F Four Region 6 assemblies (6). Two storage cells contain a stainless steel L-insert. The other two cells contain no inserts.	**6**	**6**
	6	**6**

Notes:
1. The shaded locations indicate cells which contain a stainless steel L-insert.
2. A blocked cell (X) contains a blocking device.
3. NETCO-SNAP-IN® inserts must be oriented in the same direction as the stainless steel L-inserts.
4. NETCO-SNAP-IN® inserts are only located in cells without a stainless steel L-insert.
5. Any cell containing a fuel assembly or a TC may instead be an empty (water-filled) cell in all storage arrays.
6. Any storage array location designated for a fuel assembly may be replaced with non-fissile material.
7. Interface requirements: Each cell is part of up to four 2x2 arrays and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area. ~~The site is comprised of approximately 4,050 acres. Site elevations range from 890 feet above mean sea level at the southern boundary to 1,030 feet above mean sea level at the northern boundary. The minimum distance from a containment building to the exclusion area boundary is 871 meters.~~

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies.

a. Each assembly shall consist of a matrix of fuel rods with an NRC approved cladding material with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each unit-specific COLR shall contain an identification of the fuel types and cladding material in the reactor, and the associated COLR methodologies.

b. A limited number of lead test assemblies not meeting 4.2.1.a may be placed in nonlimiting core regions. Each unit-specific COLR shall contain an identification of any lead test assemblies in the reactor.

4.2.2 Control Element Assemblies

The reactor core shall contain 76 full strength and 13 part strength control element assemblies (CEAs).

The control section for the full strength CEAs shall be either boron carbide with Alloy 625 cladding, or a combination of silver-indium-cadmium and boron carbide with Alloy 625 cladding.

The control section for the part strength CEAs shall be solid Alloy 625 slugs with Alloy 625 cladding.

(continued)

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4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.80 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 900 ppm, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR.

d. A nominal 9.5 inch center-to-center distance between adjacent storage cell locations.

e. Region 1: Fuel shall be stored in a checkerboard (two-out-of-four) storage pattern. Fuel that qualifies to be stored in Regions 1, 2, 3, or 4 in accordance with Figures 3.7.17-1, 3.7.17-2, or 3.7.17-3, may be stored in Region 1.

f. Region 2: Fuel shall be stored in a repeating 3-by-4 storage pattern in which Region 2 (two-out-of-twelve) assemblies and Region 4 (ten-out-of-twelve) assemblies are mixed as shown in Section 9.1 of the UFSAR. Only fuel that qualifies to be stored in Regions 2, 3, or 4, in accordance with Figures 3.7.17-1, 3.7.17-2, or 3.7.17-3, may be stored in Region 2.

g. Region 3: Fuel shall be stored in a four-out-of-four storage pattern. Only fuel that qualifies to be stored in Regions 3 or 4, in accordance with Figures 3.7.17-2 or 3.7.17-3, may be stored in Region 3.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.65 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 1600 ppm, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR.

d. A nominal 9.5 inch center-to-center distance between adjacent storage cell locations.

e. Fuel assemblies are classified in Fuel Regions 1-6 as shown in Tables 3.7.17-1 through 3.7.17-5.

(continued)

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4.0 DESIGN FEATURES (continued)

h. Region 4: Fuel shall be stored in a repeating 3-by-4 storage pattern in which Region 2 (two-out-of-twelve) assemblies and Region 4 (ten-out-of-twelve) assemblies are mixed as shown in Section 9.1 of the UFSAR. Only fuel that qualifies to be stored in Region 4 in accordance with Figure 3.7.17-3 shall be stored in Region 4.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.80 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 17 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

(continued)

4.0 DESIGN FEATURES (continued)

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.65 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 18 inch (east-west) and 31 inch (north-south) center-to-center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include recirculation portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, and ~~the post-accident sampling subsystem of the reactor coolant sampling system (until such time as a modification eliminates the PASS penetration as a potential leakage path),~~ the containment spray system~~, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem (until such time as a modification eliminates the PASS penetration as a potential leakage path)~~. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Deleted

(continued)

5.5 Programs and Manuals (continued)

5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

Certain storage cells in the spent fuel storage racks utilize neutron absorbing material that is credited in the spent fuel storage rack criticality safety analysis to ensure the limitations of Technical Specifications 3.7.17 and 4.3.1.1 are maintained.

In order to ensure the reliability of the neutron absorber material, a monitoring program is provided to confirm the assumptions in the spent fuel pool criticality safety analysis.

The Spent Fuel Storage Rack Neutron Absorber Monitoring Program shall require periodic inspection and monitoring of spent fuel pool test coupons and neutron absorber inserts on a performance-based frequency, not to exceed 10 years.

Test coupons shall be inspected as part of the monitoring program. These inspections shall include visual, B-10 areal density and corrosion rate.

Visual in-situ inspections of inserts shall also be part of the program to monitor for signs of degradation. In addition, an insert shall be removed periodically for visual inspection, thickness measurements, and determination of retention force.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In addition to the provisions of 10 CFR 20.1601, the following controls provide an alternate method for controlling access to high radiation areas as provided by paragraph 20.1601(c) of 10 CFR part 20. High radiation areas, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a ~~Radiation Exposure Permit (REP)~~Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technicians) or personnel continuously escorted by such individuals may be exempt from the ~~REP~~RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by radiation protection supervision, or as designated in the RWP~~the Radiation Protection Section Leader or designated alternate in the REP~~.

(continued)

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Manager on duty or ~~Radiation Protection~~ radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved ~~REP~~RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the ~~REP~~RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose in excess of 1000 mrem (measurement made at 30 cm from source of radioactivity), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

ATTACHMENT 2:

Proposed Technical Specification Changes (Clean)

Changed Page(s)

3.1.5-3
3.7.17-1
3.7.17-2
3.7.17-3
3.7.17-4
3.7.17-5
3.7.17-6
3.7.17-7
4.0-1
4.0-2
4.0-3
5.5-2
5.5-22
5.7-1
5.7-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify the indicated position of each full strength and part strength CEA is within 6.6 inches of all other CEAs in its group.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.2	Verify that, for each CEA, its OPERABLE CEA position indicator channels indicate within 5.2 inches of each other.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.3	Verify full strength CEA freedom of movement (trippability) by moving each individual full strength CEA that is not fully inserted in the core at least 5 inches.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.4	Perform a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.5.5	Verify each full strength CEA drop time $\leq$ 4.0 seconds.	Prior to reactor criticality, after each removal of the reactor head

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 The combination of initial enrichment, burnup, and decay time of each fuel assembly shall be in compliance with the requirements specified in Tables 3.7.17-1 through 3.7.17-5.

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	----------NOTE---------- LCO 3.0.3 is not applicable. ---------------------------- Initiate action to move the noncomplying fuel assembly into an appropriate region.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Tables 3.7.17-1 through 3.7.17-5, Figure 3.7.17-1, and Specification 4.3.1.1.	Prior to storing the fuel assembly in the fuel storage pool.

Table 3.7.17-1

Fuel Regions Ranked by Reactivity	
Fuel Region 1	Highest Reactivity (See Note 2)
Fuel Region 2	
Fuel Region 3	
Fuel Region 4	
Fuel Region 5	
Fuel Region 6	Lowest Reactivity

Notes:

1. Fuel Regions are defined by assembly average burnup, initial enrichment[1] and decay time as provided by Table 3.7.17-2 through Table 3.7.17-5.
2. Fuel Regions are ranked in order of decreasing reactivity, e.g., Fuel Region 2 is less reactive than Fuel Region 1, etc.
3. Fuel Region 1 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U. No burnup is required.
4. Fuel Region 2 contains fuel with an initial maximum radially averaged enrichment up to 4.65 wt% ^{235}U with at least 16.0 GWd/MTU of burnup.
5. Fuel Regions 3 through 6 are determined from the minimum burnup (BU) equation and coefficients provided in Tables 3.7.17-2 through 3.7.17-5.
6. Assembly storage is controlled through the storage arrays defined in Figure 3.7.17-1.
7. Each storage cell in an array can only be populated with assemblies of the Fuel Region defined in the array definition or a lower reactivity Fuel Region.

[1]Initial Enrichment is the nominal ^{235}U enrichment of the central zone region of fuel, excluding axial blankets, prior to reduction in ^{235}U content due to fuel depletion. If the fuel assembly contains axial regions of different ^{235}U enrichment values, such as axial blankets, the maximum initial enrichment value is to be utilized.

Table 3.7.17-2

Fuel Region 3: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	-0.8100	6.5551	-2.9050	-21.0499
5	-0.9373	7.6381	-6.0246	-18.0299
10	-0.8706	6.8181	-3.1913	-21.0299
15	-0.7646	5.6311	0.7657	-25.1599
20	-0.7233	5.1651	2.3084	-26.7499

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 2.50 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 2.50 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Table 3.7.17-3

Fuel Region 4: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.0333	-2.1141	27.4985	-41.8258
5	-0.2105	0.2472	19.7919	-34.2641
10	0.0542	-2.5298	28.0953	-41.7092
15	0.3010	-5.0718	35.6966	-48.5494
20	0.4829	-6.9436	41.3118	-53.6182

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the "minimum burnup" (GWd/MTU) given by the curve fit for the assembly "decay time" and "initial enrichment." The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.75 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.75 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Table 3.7.17-4

Fuel Region 5: Burnup Requirement Coefficients				
Decay Time (yr.)	Coefficients			
	A_1	A_2	A_3	A_4
0	0.1586	-3.0177	28.7074	-39.8636
5	-0.2756	1.3433	14.5578	-26.4388
10	-0.2897	1.3218	14.6176	-26.4160
15	-0.0736	-0.9107	21.2118	-32.1887
20	0.1078	-2.7684	26.6911	-36.9873

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.65 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.65 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Table 3.7.17-5

Fuel Region 6: Burnup Requirement Coefficients				
Decay Time (yr.)	**Coefficients**			
	A_1	**A_2**	**A_3**	**A_4**
0	0.4890	-6.7447	42.7619	-49.3143
5	0.5360	-6.9115	41.1003	-46.6977
10	0.4779	-6.1841	37.6389	-43.0309
15	0.4575	-5.8844	35.8656	-41.0274
20	0.3426	-4.7050	31.8126	-37.2800

Notes:

1. Relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Region, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “decay time” and “initial enrichment.” The specific minimum burnup (BU) required for each fuel assembly is calculated from the following equation:

$$BU = A_1 * En^3 + A_2 * En^2 + A_3 * En + A_4$$

2. Initial enrichment, En, is the maximum radial average ^{235}U enrichment. Any En value between 1.45 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 1.45 wt% ^{235}U.

3. It is acceptable to linearly interpolate between calculated BU limits based on decay time.

4. The 20-year coefficients must be used to calculate the minimum BU for an assembly with a decay time of greater than 20 years.

Figure 3.7.17-1
Allowable Storage Arrays

<table>
<tr><td rowspan="2">Array A
Two Region 1 assemblies (1) checkerboarded with two blocked cells (X). The Region 1 assemblies are each in a cell with a stainless steel L-insert. No NETCO-SNAP-IN® inserts are credited.</td><td>1</td><td>X</td></tr>
<tr><td>X</td><td>1</td></tr>
<tr><td rowspan="2">Array B
Two Region 1 assemblies (1) checkerboarded with two cells containing trash cans (TC). The Region 1 assemblies are each in a cell with a stainless steel L-insert. Every cell without a stainless steel L-insert must contain a NETCO-SNAP-IN® insert.</td><td>1</td><td>TC</td></tr>
<tr><td>TC</td><td>1</td></tr>
<tr><td rowspan="2">Array C
Two Region 2 assemblies (2) checkerboarded with one Region 3 assembly (3) and one blocked cell (X). The Region 2 assemblies are each in a cell with a stainless steel L-insert. The Region 3 assembly is in a cell containing a NETCO-SNAP-IN® insert.</td><td>2</td><td>X</td></tr>
<tr><td>3</td><td>2</td></tr>
<tr><td rowspan="2">Array D
One Region 2 assembly (2) checkerboarded with three Region 4 assemblies (4). The Region 2 assembly and the diagonally located Region 4 assembly are each in a storage cell with a stainless steel L-insert. The two storage cells without a stainless steel L-insert contain a NETCO-SNAP-IN® insert.</td><td>2</td><td>4</td></tr>
<tr><td>4</td><td>4</td></tr>
<tr><td rowspan="2">Array E
Four Region 5 assemblies (5). Two storage cells contain a stainless steel L-insert. One cell contains a NETCO-SNAP-IN® insert. One storage cell contains no insert.</td><td>5</td><td>5</td></tr>
<tr><td>5</td><td>5</td></tr>
<tr><td rowspan="2">Array F
Four Region 6 assemblies (6). Two storage cells contain a stainless steel L-insert. The other two cells contain no inserts.</td><td>6</td><td>6</td></tr>
<tr><td>6</td><td>6</td></tr>
</table>

Notes:
1. The shaded locations indicate cells which contain a stainless steel L-insert.
2. A blocked cell (X) contains a blocking device.
3. NETCO-SNAP-IN® inserts must be oriented in the same direction as the stainless steel L-inserts.
4. NETCO-SNAP-IN® inserts are only located in cells without a stainless steel L-insert.
5. Any cell containing a fuel assembly or a TC may instead be an empty (water-filled) cell in all storage arrays.
6. Any storage array location designated for a fuel assembly may be replaced with non-fissile material.
7. Interface requirements: Each cell is part of up to four 2x2 arrays and each cell must simultaneously meet the requirements of all those arrays of which it is a part.

4.0 DESIGN FEATURES

4.1 Site Location

The Palo Verde Nuclear Generating Station is located in Maricopa County, Arizona, approximately 50 miles west of the Phoenix metropolitan area.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 241 fuel assemblies.

a. Each assembly shall consist of a matrix of fuel rods with an NRC approved cladding material with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each unit-specific COLR shall contain an identification of the fuel types and cladding material in the reactor, and the associated COLR methodologies.

b. A limited number of lead test assemblies not meeting 4.2.1.a may be placed in nonlimiting core regions. Each unit-specific COLR shall contain an identification of any lead test assemblies in the reactor.

4.2.2 Control Element Assemblies

The reactor core shall contain 76 full strength and 13 part strength control element assemblies (CEAs).

The control section for the full strength CEAs shall be either boron carbide with Alloy 625 cladding, or a combination of silver-indium-cadmium and boron carbide with Alloy 625 cladding.

The control section for the part strength CEAs shall be solid Alloy 625 slugs with Alloy 625 cladding.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.65 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 1600 ppm, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR.

d. A nominal 9.5 inch center-to-center distance between adjacent storage cell locations.

e. Fuel assemblies are classified in Fuel Regions 1-6 as shown in Tables 3.7.17-1 through 3.7.17-5.

(continued)

4.0 DESIGN FEATURES (continued)

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum radially averaged U-235 enrichment of 4.65 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for biases and uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 18 inch (east-west) and 31 inch (north-south) center-to-center distance between fuel assemblies placed in the storage racks.

4.3.2 <u>Drainage</u>

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 137 feet - 6 inches.

4.3.3 <u>Capacity</u>

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1329 fuel assemblies.

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include recirculation portion of the high pressure injection system, the shutdown cooling portion of the low pressure safety injection system, and the containment spray system. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Deleted

(continued)

5.5 Programs and Manuals (continued)

5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

Certain storage cells in the spent fuel storage racks utilize neutron absorbing material that is credited in the spent fuel storage rack criticality safety analysis to ensure the limitations of Technical Specifications 3.7.17 and 4.3.1.1 are maintained.

In order to ensure the reliability of the neutron absorber material, a monitoring program is provided to confirm the assumptions in the spent fuel pool criticality safety analysis.

The Spent Fuel Storage Rack Neutron Absorber Monitoring Program shall require periodic inspection and monitoring of spent fuel pool test coupons and neutron absorber inserts on a performance-based frequency, not to exceed 10 years.

Test coupons shall be inspected as part of the monitoring program. These inspections shall include visual, B-10 areal density and corrosion rate.

Visual in-situ inspections of inserts shall also be part of the program to monitor for signs of degradation. In addition, an insert shall be removed periodically for visual inspection, thickness measurements, and determination of retention force.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In addition to the provisions of 10 CFR 20.1601, the following controls provide an alternate method for controlling access to high radiation areas as provided by paragraph 20.1601(c) of 10 CFR part 20. High radiation areas, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but ≤ 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Radiation Protection Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by radiation protection supervision, or as designated in the RWP.

(continued)

5.7 High Radiation Area

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Manager on duty or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3 For individual high radiation areas accessible to personnel with radiation levels such that an individual could receive in 1 hour a dose in excess of 1000 mrem (measurement made at 30 cm from source of radioactivity), that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

ATTACHMENT 3:

Revised Technical Specification Bases Changes (Mark-up – For Information)

Changed Page(s)

B 3.7.15-1
B 3.7.15-2
B 3.7.15-1a
B 3.7.15-2a
B 3.7.17-1
B 3.7.17-2
B 3.7.17-3
B 3.7.17-4
B 3.7.17-5
B 3.7.17-6
B 3.7.17-1a
B 3.7.17-2a
B 3.7.17-3a

Delete page

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

As described in LCO 3.7.17, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 2150 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron. In order to maintain the spent fuel pool $k_{eff} < 1.0$, a soluble boron concentration of 900 ppm is required to maintain the spent fuel pool $k_{eff} \leq 0.95$ assuming the most limiting single fuel mishandling accident.

APPLICABLE SAFETY ANALYSES

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.17 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack or between a rack and the pool walls. These incidents could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by these postulated accident scenarios.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool in order to comply with the TS 4.3.1.1.c design requirement that $k_{eff} \leq 0.95$.

(continued)

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BASES

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.1.2.
2. PVNGS Operating License Amendments 82, 69 and 54 for Units 1, 2 and 3, respectively, and associated NRC Safety Evaluation dated September 30, 1994.
3. 13-N-001-1900-1221-1, "Palo Verde Spent Fuel Pool Criticality Analysis," ABB calculation A-PV-FE-0106, revision 3, dated January 15, 1999.

B 3.7 PLANT SYSTEMS

B 3.7.15 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

As described in LCO 3.7.17, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment, discharge burnup, and decay time. A soluble boron concentration of 1600 ppm is required to maintain the spent fuel pool $k_{eff} \leq 0.95$ assuming the most limiting fuel mishandling accident.

APPLICABLE SAFETY ANALYSES

A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.17 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). There could also be a misload of multiple fuel assemblies into fuel rack locations not allowed by LCO 3.7.17. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded fuel pool storage rack or between a rack and the pool walls. These incidents could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by these postulated accident scenarios.

The concentration of dissolved boron in the fuel pool satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The specified concentration of dissolved boron in the fuel pool preserves the assumptions used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel pool.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool in order to comply with the TS 4.3.1.1.c design requirement that $k_{eff} \leq 0.95$.

(continued)

BASES

ACTIONS

A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel pool is less than required, immediate action must be taken to preclude an accident from happening or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. This does not preclude the movement of fuel assemblies to a safe position. In addition, action must be immediately initiated to restore boron concentration to within limit.

If moving fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies that the concentration of boron in the spent fuel pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 9.1.2.

2 “Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3” (Proprietary), WCAP-18030-P, Revision 1, October 2016.

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B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool was originally designed to store up to 1329 fuel assemblies in a borated fuel storage mode. The current storage configuration, which allows credit to be taken for boron concentration, burnup, and decay time, and does not require neutron absorbing (boraflex) storage cans, provides for a maximum storage of 1209 fuel assemblies in a four-region configuration. The design basis of the spent fuel cooling system, however, is to provide adequate cooling to the spent fuel during all operating conditions (including full core offload) for only 1205 fuel assemblies (UFSAR section 9.1.3). Therefore, an additional four spaces are mechanically blocked to limit the maximum number of fuel assemblies that may be stored in the spent fuel storage pool to 1205.

Region 1 is comprised of two 9x8 storage racks and one 12x8 storage rack. To prevent inadvertent storage of a fuel assembly in a cell required to be vacant, cell blocking devices are placed in every other storage cell location in Region 1 to maintain a two-out-of-four checkerboard configuration.

Region 3 is comprised of three 9x8 storage racks and one 9x9 storage rack in Units 2 and 3. Region 3 is comprised of four 9x8 storage racks and one 9x9 storage rack in Unit 1. Since fuel assemblies may be stored in every Region 3 cell location, no cell blocking devices are installed in Region 3.

Regions 2 and 4 are mixed and are comprised of seven 9x8 storage racks and three 12x8 storage racks in Units 2 and 3, Regions 2 and 4 are mixed and are comprised of six 9x8 storage racks and three 12x8 storage racks in Unit 1. Regions 2 and 4 are mixed in a repeating 3x4 storage pattern in which two-out-of-twelve cell locations are designated Region 2 and ten-out-of-twelve cell locations are designated Region 4 (see UFSAR Figures 9.1-7 and 9.1-7A). Since fuel assemblies may be stored in every Region 2 and Region 4 cell location, no cell blocking devices are installed in Region 2 and Region 4.

(continued)

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BASES

BACKGROUND (continued)

The spent fuel storage cells are installed in parallel rows with a nominal center-to-center spacing of 9.5 inches. This spacing, a minimum soluble boron concentration of 900 ppm, and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 is sufficient to maintain a k_{eff} of $\leq$0.95 for fuel of original maximum radially averaged enrichment of up to 4.80%.

APPLICABLE SAFETY ANALYSES

The spent fuel storage pool is designed for non-criticality by use of adequate spacing, credit for boron concentration, and the storage of fuel in the appropriate region based on assembly burnup in accordance with TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3. The design requirements related to criticality (TS 4.3.1.1) are k_{eff} < 1.0 assuming no credit for boron and $k_{eff} \leq 0.95$ taking credit for soluble boron. The burnup versus enrichment requirements (TS Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3) are developed assuming k_{eff} < 1.0 with no credit taken for soluble boron, and that $k_{eff} \leq 0.95$ assuming a soluble boron concentration of 900 ppm and the most limiting single fuel mishandling accident.

The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed by ABB-Combustion Engineering (CE) using the three-dimensional Monte Carlo code KENO-VA with the updated 44 group ENDF/B-5 neutron cross section library. The KENO code has been previously used by CE for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the PVNGS fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment and assembly spacing.

The modeling of Regions 2, 3, and 4 included several conservative assumptions. These assumptions neglected the reactivity effects of poison shims in the assemblies and structural grids. These assumptions tend to increase the calculated effective multiplication factor (k_{eff}) of the racks. The stored fuel assemblies were modeled as CE 16x16 assemblies with a nominal pitch of 0.5065 inches between fuel rods, a fuel pellet diameter of 0.3255 inches, and a U0(2) density of 10.31 g/cc.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

KENO-Va calculations were used to construct curves of burnup versus initial enrichment for decay times in 5 year increments from 0 to 20 years for both Regions 3 and 4 (TS Figures 3.7.17-2 and 3.7.17-3) such that all points on the curves produce a k_{eff} value (including all biases and uncertainties) of < 1.0 for unborated water. Core operating conditions, such as temperature and boron concentration, influence plutonium production and may increase the discharged fuel reactivity which could impact those curves. Biases associated with methodology and water temperature were included, and uncertainties associated with methodology, KENO-Va calculation, fuel enrichment, fuel rack pitch, fuel rack and L-insert thickness, pellet stack density, and asymmetric fuel assembly loading were included. KENO-Va calculations were also performed to determine the soluble boron concentration required to maintain the spent fuel pool k_{eff} (including all biases and uncertainties) ≤ 0.95 at a 95% probability/95% confidence level. A soluble boron concentration of 900 ppm is required to assure that the spent fuel pool k_{eff} remains ≤ 0.95 at all times. This soluble boron concentration accounts for the positive reactivity effects of the most limiting single fuel mishandling event and uncertainties associated with fuel assembly reactivity and burnup. This method of reactivity equivalencing has been accepted by the NRC (Reference 3) and used for numerous other spent fuel storage pools that take credit for burnup, decay time, and soluble boron.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, with a burnup and enrichment combination outside of the acceptable area in TS Figure 3.7.17-1, or with a burnup, decay time, and enrichment combination outside of the acceptable area in TS Figures 3.7.17-2 or 3.7.17-3, which could lead to an increase in reactivity. These events would include an assembly drop on top of a rack or between a rack and the pool walls, or the misloading of an assembly. For such events, partial credit may be taken for the soluble boron in the spent fuel pool water to ensure protection against a criticality accident since the staff does not require the assumption of two unlikely, independent, concurrent events (double contingency principle). Although a soluble boron concentration of only 900 ppm is required to assure that k_{eff} remains ≤ 0.95 assuming the single most limiting fuel mishandling event, TS 3.7.15 conservatively requires the presence of 2150 ppm of soluble boron in the spent fuel pool water. As such, the reduction in k_{eff} caused

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BASES

APPLICABLE SAFETY ANALYSES (continued)

by the required soluble boron concentration more than offsets the reactivity addition caused by credible accidents accidents, and the staff criterion of $k_{eff} \leq 0.95$ is met at all times.

The criticality aspects of the spent fuel pool meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

The spent fuel pool heat load calculations were based on a full pool with 1205 fuel assemblies. From the spent fuel pool criticality analysis, the number of fuel assemblies that can be stored in the four-region configuration is 1209 fuel assemblies. The design basis of the spent fuel cooling system, however, is to provide adequate cooling to the spent fuel during all operating conditions (including full core offload) for only 1205 fuel assemblies (UFSAR section 9.1.3). Therefore, an additional four spaces are mechanically blocked to limit the maximum number of fuel assemblies that may be stored in the spent fuel storage pool to 1205.

The original licensing basis for the spent fuel pool allowed for spent fuel to be loaded in either a 4x4 array or a checkerboard array, depending on the use of borated poison. A fuel handling accident was assumed to occur with maximum loading of the pool. The fuel pool rack construction precludes more than one assembly from being impacted in a fuel handling accident. The UFSAR analysis conclusion regarding the worst scenario for a dropped assembly (in which the horizontal impact of a fuel assembly on top of the spent fuel assembly damages fuel rods in the dropped assembly but does not impact fuel in the stored assemblies) continues to be limiting.

The spent fuel assembly storage satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 1.0 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 in the accompanying LCO. Specification 4.3.1.1 provides additional details for fuel storage in each of the four Regions.

(continued)

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BASES

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figures 3.7.17-1, 3.7.17-2, and 3.7.17-3 in the accompanying LCO and Specification 4.3.1.1.

To manually determine the allowed SFP region for a fuel assembly, the actual burnup is compared to the burnup requirement for the given initial enrichment and appropriate decay time from Figure 3.7.17-1, 3.7.17-2, or 3.7.17-3. If the actual burnup is greater than or equal to the burnup requirement, then the fuel assembly is eligible to be stored in the corresponding region. If the actual burnup is less than the burnup requirement, then the comparison needs to be repeated using another curve for a lower numbered region. Note the following:

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

- that a fuel assembly that does not meet the burnup requirement for Region 2 must be stored in Region 1,
- that any fuel assembly may be stored in Region 1,
- that any fuel assembly may be stored in a lower numbered region than the region for which it qualifies because burnup requirements decrease as region numbers decrease (refer also to Tech Spec 4.3.1.1),
- and that comparing actual burnup to the burnup requirement for zero decay time will always be correct or conservative.

REFERENCES

1. UFSAR, Sections 9.1.2 and 9.1.3.
2. PVNGS Operating License Amendments 82, 69, and 54 for Units 1, 2, and 3 respectively, and associated NRC Safety Evaluation, dated September 30, 1994.
3. Letter to T. E. Collins, U.S. NRC to T. Greene, WOG, "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC NO. M93254)", October 25, 1996.
4. 13-N-001-1900-1221-1, "Palo Verde Spent Fuel Pool Criticality Analysis," ABB calculation A-PV-FE-0106, revision 03, dated January 15, 1999.
5. Westinghouse letter NF-APS-10-19, “Criticality Safety Evaluation of the Spent Fuel Pool Map with a Proposed Region 3 Increase,” dated February 25, 2010.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel pool is designed to store new (nonirradiated) fuel assemblies and burned (irradiated) fuel assemblies in a vertical configuration underwater. The storage pool was originally designed to store up to 1329 fuel assemblies. The design basis of the spent fuel pool cooling system is to provide adequate cooling to the spent fuel pool during all operating conditions (including full core offload) for up to 1205 fuel assemblies (UFSAR Section 9.1.3).

The spent fuel storage cells are installed in parallel rows with a nominal center-to-center spacing of 9.5 inches. This spacing, a minimum soluble boron concentration of 1600 ppm, the use of neutron-absorbing panels, and the storage of fuel in the appropriate region based on fuel assembly initial enrichment, discharge burnup, and decay time in accordance with TS Tables 3.7.17-1 through 3.7.17-5 is sufficient to maintain $k_{eff} \leq 0.95$ for fuel of initial maximum radially averaged enrichment of up to 4.65 wt%. To prevent inadvertent storage of a fuel assembly in a cell required to be vacant, cell blocking devices are placed in every other storage cell location in Region 1 to maintain a two-out-of-four checkerboard configuration.

Disused CEAs, in-core instruments, and other material is stored in trash cans. A trash can may be stored in any location that is approved to store a fuel assembly. No special nuclear material (SNM) may be stored in a trash can.

APPLICABLE SAFETY ANALYSES

The nuclear criticality safety analysis in References 1 and 2 considered the following reactivity-increasing accidents:

- Misload of a single assembly into an unacceptable storage location
- Multiple assemblies misloaded in series due to a common cause
- Spent fuel pool temperature outside the allowable operating range
- Dropped and misplaced fresh fuel assembly
- Seismic event
- Inadvertent removal of a NETCO-SNAP-IN® rack insert

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	In each case, the spent fuel assembly storage met the requirements of 10 CFR 50.68(b)(4). Thus, the spent fuel storage facility is designed for noncriticality by use of adequate spacing, and neutron absorbing panels considering initial enrichment, fuel burnup, and decay time. Core operating conditions, such as temperature and boron concentration, influence plutonium production and may increase the discharged fuel reactivity which could impact those curves. The spent fuel assembly storage satisfies criterion 2 of 10 CFR 50.36 (c)(2)(ii).
LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1 in the accompanying LCO, ensures that the k_{eff} of the spent fuel pool will always remain < 1.0 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent fuel pool according to Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1 in the accompanying LCO. Specification 4.3.1.1 provides additional details for fuel storage in each of the six Regions.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1.

(continued)

BASES

ACTIONS

A.1 (continued)

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies by administrative means that the initial enrichment, discharge burnup, and decay time of the fuel assembly is in accordance with Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1 in the accompanying LCO. For fuel assemblies in the unacceptable range of Tables 3.7.17-1 through 3.7.17-5, performance of this SR will ensure compliance with Specification 4.3.1.1.

REFERENCES

1. UFSAR, Sections 9.1.2 and 9.1.3.
2. "Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3" (Proprietary), WCAP-18030-P, Revision 1, October 2016.