Enclosure Attachment 8 contains PROPRIETARY information to be withheld under 10 CFR 2.390

10 CFR 50.90

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102-07149-MLL/TNW
November 25, 2015

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

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2, and 3
Docket Nos. STN 50-528, 50-529, and 50-530
License Amendment Request to Revise Technical Specifications to Incorporate Updated Criticality Safety Analysis

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), Arizona Public Service Company (APS) is submitting a request for a license amendment to revise the Technical Specifications (TS) for Palo Verde Nuclear Generating Station Units 1, 2, and 3. The proposed amendment would modify TS requirements to incorporate the results of an updated criticality safety analysis for both new and spent fuel storage.

The enclosure to this letter provides a description and assessment of the proposed changes including a technical evaluation, a regulatory evaluation, a significant hazards consideration, and an environmental consideration. The enclosure also contains eight attachments. Attachment 1 provides the marked-up existing TS pages. Attachment 2 provides the revised (clean) TS pages. Attachment 3 provides the marked-up TS Bases pages to show the proposed changes.

This submittal contains new regulatory commitments (as defined by NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, Revision 0) to be implemented, which are identified in Attachment 4. Attachment 5 provides a non-proprietary version of the criticality safety analysis. Attachment 6 provides a material qualification report for NETCO-SNAP-IN® neutron absorbing spent fuel pool rack inserts.

Attachment 7 is an affidavit signed by Westinghouse Electric Company LLC that sets forth the basis on which the proprietary information in Attachment 8 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). Correspondence with respect to the proprietary aspects of Attachment 8

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Attachment 8 transmitted herewith contains PROPRIETARY information.
When separated from Attachment 8, this transmittal document is decontrolled.

or the supporting Westinghouse affidavit should reference Westinghouse letter number CAW-15-4271 and be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Attachment 8 is the *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3,* WCAP-18030-P (Proprietary), which contains information proprietary to Westinghouse Electric Company LLC.

A public pre-submittal meeting was held with the NRC on May 11, 2015 (Agency Document Access and Management System [ADAMS] accession number ML15140A314) to discuss the criticality safety analysis performed in support of this license amendment request. A follow-up public conference call to address action items from the May 11, 2015, pre-submittal meeting was held on September 1, 2015 (ADAMS accession number ML15286A028).

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

APS requests approval of the proposed license amendment by October 1, 2017, and will implement the TS amendment within 90 days following NRC approval. This request is necessary to complete the Spent Fuel Pool Transition Plan by the end of 2019.

Should you have any questions concerning the content of this letter, please contact Thomas Weber, Department Leader, Nuclear Regulatory Affairs, at (623) 393-5764.

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

Executed on November 25, 2015
(Date)

Sincerely,

Mark Lacal

MLL/TNW/JR/af

Enclosure: Description and Assessment of Proposed License Amendment

cc:	M. L. Dapas	NRC Region IV Regional Administrator
	M. M. Watford	NRC NRR Project Manager for PVNGS
	L. J. Kloss	NRC NRR Project Manager
	C. A. Peabody	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)
	T. Morales	Arizona Radiation Regulatory Agency (ARRA)

Enclosure

Description and Assessment of Proposed License Amendment

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LIST OF ACRONYMS

ANP	AREVA PVNGS Lead Test Assembly Combustion Engineering 16x16 Fuel
APS	Arizona Public Service Company
ENDF	Evaluated Nuclear Data File
FHE	Fuel Handling Equipment
IFSR	Intermediate Fuel Storage Rack
LAR	License Amendment Request
LER	Licensee Event Report
NFS	New Fuel Storage
NGF	Combustion Engineering 16x16 Next Generation Fuel
PVNGS	Palo Verde Nuclear Generating Station
SFP	Spent Fuel Pool
STD	Standard Combustion Engineering 16x16 Fuel
TS	Technical Specification(s)
VAP	Value Added Pellet Combustion Engineering 16x16 Fuel

1.0 SUMMARY DESCRIPTION

The proposed amendment would revise Palo Verde Nuclear Generating Station (PVNGS) Renewed Operating License Nos. NPF-41, NPF-51, and NPF-74 to amend the Technical Specifications (TS) to incorporate the results of the updated criticality safety analysis, WCAP-18030-P (Reference 6.1). The proposed amendment will correct a non-conservative TS regarding NRC approved License Amendment Number 125, which describes the current licensing bases for the criticality safety analysis for PVNGS. This is further discussed in Section 2.2 of this proposed amendment.

This enclosure provides a description and assessment of the proposed changes including a technical evaluation, a regulatory evaluation, a significant hazards consideration, and an environmental consideration. The enclosure also contains eight attachments. Attachment 1 provides the marked-up existing TS pages. Attachment 2 provides the revised (clean) TS pages. Attachment 3 provides the marked-up TS Bases pages to show the proposed changes.

This submittal contains new regulatory commitments (as defined by NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, Revision 0) to be implemented, which are identified in Attachment 4. Attachment 5 provides a non-proprietary version of the criticality safety analysis. Attachment 6 provides a material qualification report for NETCO-SNAP-IN® neutron absorbing spent fuel pool rack inserts.

Attachment 7 is an affidavit signed by Westinghouse Electric Company LLC that sets forth the basis on which the proprietary information in Attachment 8 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4).

Attachment 8 is the *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3,* WCAP-18030-P (Proprietary), which contains information proprietary to Westinghouse Electric Company LLC.

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes to the Technical Specifications

The following specific TS changes are proposed as part of the updated criticality safety analysis. Marked-up TS pages are provided in Attachment 1 and revised (clean) TS pages are provided in Attachment 2.

- TS 3.7.17, *Spent Fuel Assembly Storage*
  - Revise the LCO statement and surveillance requirement statement to reflect the updated spent fuel assembly storage requirements resulting from WCAP-18030-P
  - Add Tables 3.7.17-1 through 3.7.17-5 to define the Fuel Regions
  - Replace Figure 3.7.17-1 to define the Allowable Storage Arrays
  - Delete Figures 3.7.17-2 and 3.7.17-3 (information is replaced by Tables 3.7.17-1 through 3.7.17-5)

- TS 4.3.1, *Criticality*
  - Change TS 4.3.1.1.a to read "4.65 weight percent" instead of "4.80 weight percent"
  - Change TS 4.3.1.1.c to read "1460 ppm" [parts per million] instead of "900 ppm"
  - Change TS 4.3.1.1.e to refer to Fuel Regions 1 – 6 as shown in Tables 3.7.17-1 through 3.7.17-5
  - Delete TS 4.3.1.1.f through 4.3.1.1.h (Fuel Regions are defined in Tables 3.7.17-1 through 3.7.17-5)
  - Change TS 4.3.1.2.a to read "4.65 weight percent" instead of "4.80 weight percent"
  - Change TS 4.3.1.2.d to replace "A nominal 17 inch center to center..." with "A nominal 18 inch (east-west) and 31 inch (north-south) center-to-center..."

- Add new program TS 5.5.21, *Spent Fuel Storage Rack Neutron Absorber Monitoring Program* (Proposed TS 5.5.20, *Risk Informed Completion Time Program*, was submitted on July 31, 2015 [Agency Document Access and Management System (ADAMS) accession number ML15218A300])

The TS Bases will also be revised for consistency with the proposed TS changes and with WCAP-18030-P. A markup of the TS Bases pages reflecting these changes is provided in Attachment 3 for information. The proposed TS Bases changes will be implemented in accordance with TS 5.5.14, *Technical Specifications (TS) Bases Control Program*, at the same time that the TS changes in the approved license amendment request (LAR) are implemented.

2.2 Need for Proposed Changes

In March of 2000, the NRC approved License Amendment Number 125, which describes the current licensing bases for the criticality safety analysis for PVNGS. That amendment increased the storage capacity of the spent fuel pools (SFPs) by allowing credit for soluble boron and decay time in the criticality safety analysis. The amendment also increased the maximum radially averaged fuel enrichment from 4.3 weight percent U-235 to 4.8 weight percent U-235. The methodology that was the basis for that amendment was analogous to that developed in WCAP-14416-P-A, *Westinghouse Spent Fuel Rack Criticality Analysis Methodology,* which was reviewed and approved by the NRC for use [NRC Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)," dated October 25, 1996]. In 2001 and 2004, Arizona Public Service Company (APS) submitted license amendment requests (LARs) to the NRC that would support replacement of the steam generators and authorize subsequent operation at an increased maximum power level of 3990 Megawatts thermal (a 2.94 percent increase). The NRC approved the amendments for Unit 2 in 2003 (ADAMS accession number ML032720538) and for Units 1 and 3 in 2005 (ADAMS accession number ML053130275).

In May of 2013, APS submitted Licensee Event Report (LER) 2013-001-00 (ADAMS accession number ML13133A002), which reported that certain impacts to the SFP criticality safety analysis approved in License Amendment 125 had not been considered by APS during the increase in the maximum power level to 3990 MWt. One of the corrective actions in the LER was to revise the SFP criticality safety analysis using updated methodology and input parameters and to

submit a LAR to correct the non-conservative TS. The criticality safety analysis methodology included in this LAR is based upon the most recent NRC approved guidance of *Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,* DSS-ISG-2010-01 (Reference 6.2) and satisfies the actions stipulated in the PVNGS Corrective Action Program and PVNGS LER 2013-001-00.

3.0 TECHNICAL EVALUATION

This LAR documents an updated criticality safety analysis for the PVNGS SFPs, new fuel storage (NFS) racks, interim fuel storage rack (IFSR) and fuel handling equipment (FHE). Attachment 8 is the plant-specific Westinghouse WCAP-18030-P, Revision 0, *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2 and 3.* A plant-specific NETCO-SNAP-IN® material qualification report is also included (Attachment 6) since the LAR credits the presence of neutron poisons in the NETCO-SNAP-IN® neutron absorber inserts. The main body of the LAR includes descriptions and summary evaluations, while the attached WCAP and NETCO report provide additional details on topics that include computer codes, fuel design history, depletion analysis, criticality analysis, as well as the interface, normal, and accident conditions for PVNGS.

The change to TS 4.3.1.2.d regarding NFS rack spacing is proposed to more accurately reflect the NFS as-built drawings and the existing NFS criticality safety analysis of record. The text of TS 4.3.1.2 states "The new fuel storage racks are designed and shall be maintained with...", which refers to physical dimensions. Therefore, TS 4.3.1.2.d must accurately reflect the as-built dimensions that must be maintained throughout the life of the plant. The racks have a nominal 18-inch center-to-center pitch on the short axis (east-west) and a 31-inch center-to-center pitch on the long axis (north-south). According to both the existing NFS and updated criticality safety analyses, this rack design maintains k_{eff} <0.95 during all normal and accident conditions.

3.1 Spent Fuel Pool Analysis

Design Approach

The existing SFP storage racks are evaluated for the placement of fuel within the storage arrays described in the proposed TS changes. Credit is taken for the negative reactivity associated with burnup and post-irradiation cooling time (decay time). Additionally, some SFP storage arrays credit the presence of the neutron poison in the NETCO-SNAP-IN® inserts. Finally, credit is taken for the presence of soluble boron in the SFP for specific conditions.

Compliance for the SFP is demonstrated by establishing limits on the minimum allowable burnup as a function of initial enrichment and decay time for each fuel storage array. A conservative combination of best estimate and bounding values has been selected to model the fuel in the analysis to ensure that fuel represented by the proposed TS is less reactive than the fuel modeled in the analysis. Therefore, burnup limits will conservatively bound fuel to be stored in the SFP.

Acceptance Criteria

The objective of the SFP criticality safety analysis is to ensure that the SFP operates within the bounds of 10 CFR 50.68(b)(4):

- If no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.
- If credit is taken for soluble boron, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k_{eff} must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Computer Codes

The SFP criticality safety analysis methodology employs the following computer codes and cross-section libraries:

- SCALE 6.1.2 (Reference 6.3) with the 238-group cross-section library based on Evaluated Nuclear Data File (ENDF)/B-VII.
- The two-dimensional transport lattice code PARAGON (Reference 6.4) and its 70-group cross-section library based on ENDF/B-VI.3

PARAGON is used for simulation of in-reactor fuel assembly depletion to generate isotopics for burnup credit applications in the SFP. PARAGON is generically approved for depletion calculations (Reference 6.4) and has been chosen for this spent fuel criticality safety analysis because it has all the attributes needed for burnup credit applications. There are no Safety Evaluation Report limitations for the use of PARAGON in UO_2 criticality analyses.

Additional discussion of the computer codes is provided in Section 2.3 of Attachment 8.

Code Validation Process

The validation of the ENDF/B-VII library with the SCALE 6.1.2 CSAS5 module is documented in Appendix A of Attachment 8. The code validation shows that SCALE 6.1.2 is an accurate tool for calculation of k_{eff} for the applications in this LAR. The benchmark calculations utilize the same computer platform and cross-section libraries that are used for the design basis calculations.

3.1.1 Fuel and Fuel Storage Descriptions

Four fuel designs were considered for the criticality safety analysis:

- Standard Combustion Engineering 16x16 (STD)
- Value Added Pellet Combustion Engineering 16x16 (VAP)
- Next Generation Fuel Combustion Engineering 16x16 (NGF)
- AREVA PVNGS Lead Test Assembly Combustion Engineering 16x16 (ANP)

Section 3.1.2 of Attachment 8 discusses the non-mechanical fuel features which are important to criticality safety and how they impact the number of distinct fuel designs to be considered in the analysis.

PVNGS has three SFPs, one in each unit, that are identical in layout. Each SFP contains a single rack design and each SFP is surrounded by a concrete wall with a stainless steel liner. The presence of neither the SFP concrete wall nor liner is credited in the criticality safety analysis. All storage arrays are conservatively assumed to be radially infinite.

The SFP storage racks are made up of individual modules, each of which is an array of fuel storage cells. The storage racks are comprised of 17 modules: twelve 8x9 arrays, four 8x12 arrays, and one 9x9 array. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells compatible with fuel assembly materials and the spent fuel borated water environment. The fuel assembly spacing of a nominal 9.5 inches center-to-center distance between adjacent storage cell locations is a minimum value after allowances are made for rack fabrication tolerances and predicted deflections resulting from a safe shutdown earthquake.

The storage racks are designed to maximize the number of storage cells available (minimize storage cell pitch) to be used in a checkerboard pattern of fresh (i.e., new) fuel and empty storage locations. Trashcans in the SFP store various non-fissile materials, such as discarded control element assemblies and in-core instrumentation tubes, filters, and reconstitution materials.

The SFP racks contain a stainless steel L-insert in every other cell location as shown in PVNGS Updated Final Safety Analysis Report (UFSAR) Figure 9.1-5 to help center the new fuel assemblies within this space. The stainless steel L-inserts are offset 11/16 inch from the cell wall. This reduces the positional uncertainty or the eccentric loading positions of fuel in the racks. There is not currently a surveillance program for the stainless steel L-inserts and one will not be included as part of this submittal for the following reasons:

- The industry has a large body of operating experience with stainless steel in SFP environments in which nothing suggests the stainless steel L-inserts would become incapable of performing their function.
- The stainless steel L-inserts are analogous to the SFP racks in their function and there is no precedent for requiring a surveillance program of the racks themselves.

A NETCO-SNAP-IN® rack insert will not be installed in a cell that contains a stainless steel L-insert. The NETCO-SNAP-IN® rack inserts are described in Attachment 6.

3.1.2 Depletion Analysis

The depletion analysis is a vital part of any SFP criticality safety analysis which uses burnup credit. The isotopic inventory of the fuel as a function of burnup is generated through the depletion analysis, thus the inputs used need to be carefully considered. Section 4 of Attachment 8 describes the methods used to determine the appropriate inputs for the generation of isotopic number densities to conservatively bound fuel depletion and storage.

Some of the salient points regarding the approach to the depletion analysis include:

- The isotopic number densities generated by the fuel depletion calculations are differentiated by fuel enrichment and decay time after discharge. The fuel has isotopic number densities which are calculated at enrichments of 3.0, 4.0, and 5.0 weight percent U-235 and decay times of 0, 5, 10, 15, and 20 years.
- The soluble boron concentration in the reactor during operation impacts the reactivity of fuel being discharged to the SFP. Boron is a strong thermal neutron absorber and its presence hardens the neutron energy spectrum in the core, creating more plutonium. It is important to account for the presence of soluble boron during reactor operation to ensure this impact is adequately accounted for in the isotopic generation.
- The fuel temperature during operation impacts the reactivity of fuel being discharged to the SFP. Increasing fuel temperature increases resonance absorption in U-238 due to Doppler broadening which leads to increased plutonium production, increasing the reactivity of the fuel.
- The limiting distributed axial burnup profiles are used with the uniform axial burnup profile to calculate the burnup limits.
- The limiting axial moderator temperature profiles are used with axially distributed and uniform burnup profiles to calculate the isotopics used in generating the burnup limits. Selecting an appropriate moderator temperature profile is important as it impacts the moderator density and the neutron spectrum during depletion, as discussed in NUREG/CR-6665 (Reference 6.5). An appropriate moderator temperature ensures the impact of moderator density on the neutron spectral effects is bounded, conservatively biasing the isotopic inventory of the fuel.
- Burnable absorber usage has been considered for the analysis and conservative assumptions have been utilized to bound the effects of burnable absorbers on fuel isotopics. The burnable absorbers that have been used include both discrete and integral burnable absorbers.
- The PVNGS fuel management strategy uses radial enrichment zoning to control fuel rod power peaking. Individual assemblies may contain two or three different fuel rod enrichments which are used to control peaking factors. A study was performed to determine the reactivity impact of operating with radial enrichment zoning instead of uniform radial zoning.
- All four of the different fuel designs listed in Section 3.1.1 and the conditions in which those designs were operated, or are planned to be operated, were considered in the depletion analysis. It became clear that the NGF design would be limiting throughout life. Therefore, the NGF design was used to develop the isotopics used in the spent fuel reactivity calculations.
- The parameters used in the final depletion calculations include core operation parameters, fuel assembly dimensions, axial burnup profiles, and moderator temperature profiles.

3.1.3 Spent Fuel Pool Criticality Analysis

KENO is the criticality code used to determine the absolute reactivity of burned and fresh fuel assemblies loaded in storage arrays. The dimensions and tolerances of the design basis fuel assembly and the fuel storage racks are the basis for the KENO models used to determine the burnup requirements for each fuel storage array, and to confirm the safe operation of the SFP

under normal and accident conditions. The trashcan characteristics are also modeled in the criticality analysis.

Differences between fuel types include changes in fuel rod dimensions, such as pellet and cladding dimensions, and structural components, such as grid material and volumes. Each of the fuel types which have been, or are planned to be, operated at the plant were considered. The bounding fuel assembly design for the analysis has been determined as described in Section 4.3 of Attachment 8.

Burnup Limit Generation

To ensure safe operation of the PVNGS SFPs, the analysis defines fuel storage arrays which dictate where assemblies can be placed in the SFPs based on enrichment (weight percent U-235), average burnup (GWd/MTU), and decay time (years) since discharge. Each assembly in the reactor core depletes under slightly different conditions and can have a different reactivity at the same burnup. This is accounted for in the analysis by using a combination of depletion parameters that together produce a bounding isotopic inventory throughout life. Additionally, while fuel manufacturing is a very tightly controlled process, assemblies are not identical. Reactivity margin is added to the KENO reactivity calculations for the generation of burnup limits to account for manufacturing deviations.

Assembly storage is controlled by defining allowable storage arrays. An array can only be populated by assemblies of the fuel region defined in the array definition or a lower reactivity fuel region. Fuel regions are defined by assembly burnup, initial enrichment, and decay time.

Reactivity biases are known variations between the real and analyzed system, and their reactivity impact is added directly to the calculated k_{eff}. Uncertainties are random dispersions around a nominal, measured quantity. Their impact is added to the calculated k_{eff} as the square root of the sum of the squares of the uncertainties. The following biases and uncertainties are accounted for in the analysis. A detailed discussion of biases and uncertainties is provided in Section 5.2.3 of Attachment 8.

- Reactivity effect of manufacturing tolerances
- Burnup measurement uncertainty
- Depletion uncertainty
- Fission product and minor actinide worth bias
- An operational uncertainty of 0.002 Δk is an additional conservativism which is added to the conservatism inherent in the specific power histories from reactor operation
- Eccentric fuel assembly positioning
- Uncertainty in the predictive capability of SCALE 6.1.2 and the associated cross-section library
- SFP temperature bias within the allowable operating range
- Borated and unborated biases and uncertainties

Interface Modeling

Interfaces are the locations where there is a change in either the storage racks or the storage requirements of the fuel in question. At PVNGS, each SFP has a single storage rack design. Therefore, the only interfaces that exist are those between arrays within the single storage rack design and the only interface conditions that need to be addressed in the analysis are those between different fuel storage arrays. Additional details are provided in Section 5.3 of Attachment 8.

Normal Conditions Considered in the Criticality Safety Analysis

There are five major types of normal conditions beyond the storage of fuel assemblies that are addressed in the criticality safety analysis.

Type 1 conditions involve placement of components in or near the intact fuel assemblies while normally stored in the storage racks. This also includes removal and reinsertion of these components into the fuel when stored in the rack positions using specifically designed tooling. Examples include control element assemblies and guide tube inserts, such as in-core instrumentation tubes. The calculation results show that any components designed to be inserted into an assembly may be stored in a fuel assembly guide tube in the SFP.

Type 2 conditions involve evolutions where the fuel assembly is removed from the normal storage rack location for a specific procedure and returned to an allowable cell after completion of the procedure, such as fuel assembly cleaning, inspection, reconstitution, or sipping. These are bounded by the criticality analysis.

Fuel assembly reconstitution is a normal condition defined as either pulling damaged fuel pins out of an assembly and reinserting intact pins with less reactivity than the damaged pin, or as removing undamaged pins from a damaged assembly for insertion in a new assembly. Damaged pins will be replaced with stainless steel pins or natural uranium pins. Additional information is provided in Section 5.4.2 of Attachment 8.

Type 3 conditions involve inserting components that are not intact fuel assemblies into the fuel storage rack cells. Examples include failed fuel rod baskets and miscellaneous maintenance equipment. Any components that do not contain fissile materials can replace a fuel assembly of any fuel region in one of the approved storage configurations.

Type 4 conditions include temporary installation of non-fissile components on the rack periphery. Analyses of the storage arrays contained in the criticality analysis assume an infinite array of storage cells. This assumption bounds the installation of any non-fissile components on the periphery of racks.

Type 5 conditions involve miscellaneous conditions that do not fit into the first four normal condition types. Examples include usage of fuel handling tools for their intended purpose, miscellaneous debris under the storage racks, and damaged storage cells.

Section 5.4 of Attachment 8 provides further details about normal conditions within the SFP.

Soluble Boron Credit

In accordance with 10 CFR 50.68, the criticality safety analysis ensures that the maximum calculated k_{eff}, including all biases and uncertainties, meet the k_{eff} limit of less than 1.0 (subcritical) if flooded with unborated water at a 95 percent probability, 95 percent confidence level. Additionally, the criticality safety analysis demonstrates that if the SFP is flooded with borated water, k_{eff} does not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

The minimum soluble boron concentration in the SFP to maintain $k_{eff} \leq 0.95$ for the limiting normal condition including biases, uncertainties, and administrative margin is 450 ppm. During normal operation, TS 3.7.15 requires a soluble boron concentration of $\geq$ 2150 ppm, but the SFP boron concentration is maintained between 4000 and 4400 ppm in accordance with Technical Requirements Manual T3.1.104, *Borated Sources – Shutdown*, and T3.1.105, *Borated Sources - Operating*.

Consideration of Criticality Accidents in the SFP

The following reactivity-increasing accidents are considered and the analysis results are provided in Section 5.6 of Attachment 8.

- Assembly misload into the storage racks – this is the limiting accident which addresses both multiple assemblies being misloaded in series into unacceptable storage locations and the misload of a single assembly into an unacceptable storage location. A multiple assembly misload is a hypothetical accident where assemblies are misloaded in series due to a common cause. A single assembly misload requires 1100 ppm of boron to maintain $k_{eff} \leq 0.95$. A multiple assembly misload requires 1460 ppm of boron to maintain $k_{eff} \leq 0.95$.
- Spent fuel temperature outside operating range - the SFP is to be operated between 60°F and 180°F, but under accident conditions this temperature could be higher.
- Dropped and misplaced fresh assembly – the analysis considers the dropping of the fuel assembly from the fuel handling machine during placement of the fuel assemblies in the racks. The dropped assembly could land horizontally on top of the other fuel assemblies in the rack. Additionally, the analysis considers the possibility to misplace a fuel assembly in a location not intended for fuel.
- Seismic event - the SFP racks are seismic category I, designed and built to withstand the maximum potential earthquake stresses in this geographic area. Section 5.6.4 of Attachment 8 provides additional details.
    - The spent fuel pool racks were originally designed to contain a 188 lb. neutron poison insert in every cell. These neutron poison inserts were never installed. As the mass of the original design is greater than the mass of the NETCO-SNAP-IN®, the original analysis bounds the proposed change.
- Inadvertent removal of a NETCO-SNAP-IN® rack insert – this is a potential reactivity-increasing accident added by the incorporation of NETCO-SNAP-IN® rack inserts. The absence of an insert will cause a reactivity increase due to the loss of neutron absorbing material from the storage array.

Fuel used to date at PVNGS has an initial radially averaged enrichment of < 4.55 weight percent. Limiting the maximum radially averaged enrichment to 4.65 weight percent mitigates the consequences of a multiple fuel assembly misload event without impacting operational flexibility.

There is no source of water within the fuel building that could reduce the boron concentration of the spent fuel pool from the value of 2150 ppm (Technical Specification LCO 3.7.15) to 1460 ppm. A fire in the fuel building at elevation 140-ft. is the limiting event for boron dilution and it bounds all normal, seismic, and pipe break scenarios. The current boron dilution analysis demonstrates that the limiting boron dilution event, which is fighting a hypothetical fire on the 140-ft level of the fuel building, will reduce the boron concentration from the TS limit of 2150 ppm to 1900 ppm. This leaves adequate margin to the 1460 ppm credited by the SFP criticality safety analysis.

3.1.4 NETCO-SNAP-IN® Rack Inserts

This proposed change would credit NETCO-SNAP-IN® rack inserts for criticality control in individual SFP storage rack cells to ensure that the requirements of TS 3.7.17 and the associated WCAP-18030-P are maintained. The NETCO-SNAP-IN® rack inserts are credited in both the borated and unborated conditions. The installation of the NETCO-SNAP-IN® rack inserts will be controlled as a design change implemented under the provisions of 10 CFR 50.59, *Changes, Tests and Experiments*, from a structural, seismic, and thermal-hydraulic perspective.

Attachment 6 describes the NETCO-SNAP-IN® rack inserts, including their manufacture, an engineering evaluation, and corrosion testing information.

<u>Neutron Absorber Monitoring Program (TS 5.5.21)</u>

Arizona Public Service will institute a performance-based long-term surveillance program for the NETCO-SNAP-IN® inserts based on manufacturer recommendations, current industry operating experience, NEI guidance, and NRC safety evaluations for other plants that are using neutron absorbing inserts. The long-term surveillance program will evolve as information from these sources changes and as the data from the PVNGS-specific inspections accumulate. The surveillance program consists of periodic inspections of MAXUS® material coupons from surveillance assemblies located in the SFPs and periodic inspection of full length inserts.

Coupon Inspections

Coupons will be selected from MAXUS® production material, identical to the material used to manufacture the inserts, for periodic inspection. Individual coupons will be subjected to pre-test and post-test characterizations. As appropriate for each coupon type, coupon characterizations may include visual inspection, high resolution photography, neutron attenuation, stress relaxation, blister and pit characterizations, as well as measurement of thickness, length, width, dry weight, and density.

A surveillance assembly to which surveillance coupons are attached, also referred to as a coupon tree, will be placed in the related SFP prior to the first installation campaign of NETCO-SNAP-IN® inserts and will reside there to support the monitoring program. Periodically, coupons

will be removed and sent to a qualified laboratory for testing. The coupon trees in the related SFP will contain 48 general coupons, 24 galvanic couple coupons, and 24 bend coupons as described below. They will be situated near the center of the active fuel region to maximize exposure from the surrounding fuel. These coupons will be monitored for changes to their physical properties and for changes to their effective areal density or signs of corrosion, which could indicate neutron absorber material degradation.

The frequency for coupon removal and inspection is shown in Table 1.

Table 1 - Frequency for Coupon Removal from Racks

Coupon Type	First Ten Years	After 10 Years with Acceptable Performance
General	2 coupons every 2 years	2 coupons every 4 years
Bend	1 coupon every 2 years	1 coupon every 4 years
Galvanic couples - 304L stainless Zircaloy Inconel 718	 1 couple every 6 years 1 couple every 6 years 1 couple every 6 years	

General and Galvanic Coupons

The general coupons in each SFP are designed to carry the largest number of performance indicators for the insert material. They will be subject to pre-examination, post-examination, and acceptance testing in accordance with Table 2.

The galvanic couple coupons are composed of a MAXUS® material coupon placed in contact with Zircaloy, Inconel 718, or 304 stainless steel. Eight of each type will be used to produce a total of 24 galvanic couples per coupon tree assembly in each SFP that will be subject to the inspections listed in Table 2.

Bend Coupons

Once installed, the NETCO-SNAP-IN® rack inserts assume a constant strain condition within the SFP storage rack cell. This compression leads to internal stresses, especially at the bend, that might make the rack inserts susceptible to stress corrosion cracking. An examination of the literature on the subject indicates in general, that high-purity aluminum and low-strength aluminum alloys are not susceptible to stress corrosion cracking. However, the surveillance bend coupons placed in the related SFP will be maintained under the same strain conditions as the inserts to provide an indication of unexpected crack phenomena. These coupons will be held in capsules that compress them from their initial manufactured bend angle to an angle of approximately 90 degrees. Table 3 provides the inspection details.

Over time, the MAXUS® material is expected to release some of the strain built up during the installation process. The material has a metal matrix core made from 1000 series aluminum and boron carbide powder with an outer clad made from 5052 aluminum. Existing literature for 1100 series aluminum shows a stress relaxation rate of 58 percent over a period of 20 years. Given that 1100 series aluminum is a softer metal than 5052 aluminum, this rate is considered conservative for the MAXUS® material due to the 5052 cladding. The acceptance criterion for stress relaxation is 60 percent over a 20-year period. This rate will be used when determining minimum retention force requirements for the inserts during installation that will still hold the inserts in place during a seismic event after relaxation has occurred.

The bend coupon capsules will be removed from the coupon tree and sent to a qualified laboratory for testing where the coupons will be removed, thus relieving the strain on the coupons and allowing them to return to an angle greater than 90 degrees. The change in internal stress can be correlated to the change in bend angle the coupon forms once it is removed from the strained condition. Deviation from the pre-characterized value will determine the amount of stress relaxation over the life of the coupon.

The stress relaxation rate is not linear, rather it tends to follow a logarithmic pattern. Therefore, a more significant loss of stress is expected in the first few years of exposure, but the relaxation rate becomes asymptotic over a longer period of time.

Table 2 - General and Galvanic Coupon Characterizations

Test	Pre-Characterization	Post-Characterization	Acceptance / Rejection Criteria
Visual (high resolution digital photo)	√	√	Evidence of visual indications of performance inhibitors.
Dimension	√	√	Min. thickness: 0.005 inch less than nominal thickness (excluding pit locations). Thickness change: any change of +0.010 inch / -0.004 inch (excluding pit locations). * Length change: any change of +/- 0.02 inch * Width change: any change of +/- 0.02 inch
Density	√	√	Any change of +/- 5%
Areal density	√	√	0.0156 g/cm^2 Boron-10 minimum loading

Weight loss as determined by dry weight	√	Any change of +/- 5%
Corrosion rate	√	< 0.05 mil/yr
Anomaly characterization	√ **	To be determined at the time of analysis.

* Length and width changes are not applicable for galvanic coupons.

** At the presence of anomalies

Table 3 - Bend Coupon Characterizations

Test	Pre-Characterization	Post-Characterization	Acceptance / Rejection Criteria
Visual (high resolution digital photo)	√	√	Evidence of visual indications of performance inhibitors.
Thickness	√	√	Min. thickness: 0.005 inch less than nominal thickness (excluding pit locations). Thickness change: any change of +0.010 inch / -0.004 inch (excluding pit locations).
Bending stress	√	√	Change in stress greater than a rate of 60% over 20 years *
Weight loss as determined by dry weight		√	Any change of +/- 5%
Anomaly characterization		√ **	To be determined at the time of analysis.

* Stress relaxation rate is not linear. Stress relaxation will be re-evaluated if 60% is exceeded.

** At the presence of anomalies

Full-Length Insert Inspections

The combined effects of adequate clearance and infrequent fuel assembly movement will preclude significant wear of the rack insert. However, to verify NETCO-SNAP-IN® material performance, a portion of the installed inserts will be subject to in-situ visual inspection and removal for detailed inspection of wear performance. For in-situ inspections at the frequency described in Table 4, rack inserts will be visually inspected by camera (while remaining in the storage racks) to monitor for physical deformities such as bubbling, blistering, corrosion pitting,

cracking, or flaking. Special attention shall be paid to the development of edge or corner defects.

A region of high duty spent fuel storage rack cell locations shall be identified for full insert removal and inspection. These locations will be monitored for fuel insertion and removal events to ensure that their service bounds that of the general population of storage locations. Once every 10 years, an insert will be fully removed from this region and will be inspected in accordance with Table 5. The thickness measurements at several locations along the full insert length will be compared with the as-built thickness measurements of the removed insert to verify it has sustained uniform wear over its service life. A visual inspection of the removed insert will also be performed.

Table 4 - Frequency for Full Insert Inspections

Inspection Type	First Ten Years	After 10 Years with Acceptable Performance
In-situ	2 inserts every 2 years	2 inserts every 4 years
Removal	1 insert every 10 years	

Table 5 - Full Insert Removal Inspection Characterizations

Test	Pre-Characterization	Post-Characterization	Acceptance / Rejection Criteria
Visual (high resolution digital photo)	√	√	Evidence of visual indications of performance inhibitors.
Thickness	√	√	Min. thickness: 0.005 inch less than nominal thickness (excluding pit locations). Thickness change: any change of +0.010 inch / - 0.004 inch (excluding pit locations).
Retention force		√	Retention force less than 50 lbs

3.1.5 Spent Fuel Pool Configuration Control (Human Performance Enhancements)

APS has a multi-tier defense-in-depth program to prevent and mitigate the severity of a scenario in which multiple fuel assemblies are located in the wrong storage locations. Specific aspects of this program are described below.

Control of Move Sheet Generation

- Detailed administrative procedures for the generation of move sheets and revision of move sheets
- Training and qualification of individuals responsible for generation and revision of move sheets for use and implementation of the new TS proposed in this LAR.
- Graphical representation of approved arrays in TS 3.7.17 to minimize the probability of misinterpretations
- In accordance with the plant special nuclear materials procedures, APS maintains a Region Specification Document for each SFP to aid the move sheet preparer and verifier in selecting and verifying proper placement of fuel assemblies. This document tracks the following for every fuel assembly at PVNGS:
  - Fuel assembly initial enrichment
  - Fuel assembly burnup
  - Limiting Fuel Region in which the fuel assembly can be stored
- Every fuel move is checked against the availability of the space and the eligibility of fuel to be stored there
- A member of management confirms that each move sheet was generated in accordance with PVNGS procedures. There are at least three signatures on each move sheet sent to the field, including the move sheet preparer, the verifier, and management.
- A move sheet package is a change document that is used to specify and record changes to plant configuration as it relates to special nuclear material. A move sheet contains as a minimum, the item to be moved, the "from" location, and the "to" location. Any number of fuel assemblies can be moved using a single move sheet package.

Control of Fuel Movement

- The spent fuel handling machine is only operated using approved procedures
- All individuals operating the spent fuel handling machine and acting as independent verifiers are trained in their position, including training on industry operating experience pertaining to fuel misload events
- Fuel is moved only as directed by approved move sheets
- The correct location of the spent fuel handling machine is independently verified before a fuel move begins
- The correct location of the spent fuel handing machine is independently verified before the fuel is placed in the SFP racks
- Continuous communication is maintained between the fuel mover and verifier

- The fuel handlers visually confirm that fresh fuel is not placed in locations that are "face-adjacent" to other fresh fuel assemblies

Use of Blocking Devices

The assumed limiting misload event at PVNGS involves placing a fuel assembly in a location that is required to be empty per TS 3.7.17. APS uses blocking devices to minimize the probability of a fuel assembly being placed in one of these limiting locations. Each blocking device meets the following criteria:

- Physically configured to prevent insertion of a fuel assembly in a fuel storage location
- Requires special tools to install or remove the blocking device from a storage location
- The tool used to grapple a fuel assembly is physically incapable of grappling a blocking device
- Designed to preclude falling into a storage location or becoming dislodged during normal operation
- Will support the full load of a fuel assembly and the fuel assembly grappling tool
- Allows continuous water flow through the storage cell
- Is easy to distinguish visually from a fuel assembly
- Blocking devices are administratively controlled with the same level of rigor as fuel assemblies
- A blocking device move sheet package is a change document used to specify and record changes to plant configuration as it relates to blocking devices. A blocking device move sheet contains, as a minimum, the "from" location and the "to" location of the blocking device. Any number of blocking devices may be moved using a single blocking device move sheet package. Blocking devices will not be moved using the same move sheet package as fuel assemblies. This restriction prevents a single error from removing a blocking device and placing a fuel assembly in a location that is required to be empty.

Confirmation of Configuration Control

In accordance with existing procedures, a 100 percent serial number check is performed once per calendar year to ensure that every fuel assembly stored in the SFP matches the SFP maps. This provision limits the amount of time that a misload condition could potentially exist.

Mitigation of a Misload Event

If the controls discussed above are insufficient to prevent a fuel assembly from being misloaded, the following will mitigate the consequences of such an event:

- Misload events, including misload events involving multiple fuel assemblies, have been analyzed
  - An adequate soluble boron margin mitigates the misload event. TS 3.7.15 requires 2150 ppm of soluble boron
    - The limiting misload of a single fuel assembly requires 1100 ppm of soluble boron to maintain k_{eff} <0.95

        - The limiting, credible, multiple misload of placing a fresh fuel assembly into every blocked location in the most limiting array (Array C) requires 1460 ppm of soluble boron to maintain k_{eff} <0.95
- Placing fresh fuel assemblies face-adjacent to one another is not credible. A defense-in-depth approach provides multiple, independent barriers to this event. These barriers include:
    - Move sheets are generated, independently verified, and approved by qualified individuals
    - Blocking devices or trash cans are placed in locations that are face-adjacent to locations approved for the storage of fresh fuel
    - The fuel movers will verify that fresh fuel is not placed in face-adjacent locations prior to completing each fuel move

Sufficient rigor is placed into the generation of move sheets, execution of fuel movement, and maintenance of SFP maps that the likelihood of an assembly being misplaced in the SFP is small. The misplacement of multiple fuel assemblies is less probable. Therefore, the approach used in the analysis is appropriately conservative. The impact of placing multiple fresh fuel assemblies in face-adjacent locations is not evaluated in the analysis because this event is not considered credible.

A regulatory commitment regarding the implementation of procedural controls to require verification that fresh fuel assemblies are not placed face-adjacent to one another before completing a fuel move is provided in Attachment 4.

3.2 New Fuel Storage and Fuel Transfer Equipment Analysis

A criticality safety analysis was performed to support operation of the NFS racks, the IFSR, the new fuel elevator, and the fuel upender and transfer machine. When discussing the new fuel elevator, and the fuel upender and transfer machine together, they are referred to as the fuel handling equipment (FHE). The existing NFS racks, IFSR, and FHE were evaluated to confirm that each system maintains subcriticality while performing its designed purposes.

3.2.1 Storage and Equipment Description

New Fuel Storage Design

The NFS rack assemblies are made up of individual racks similar to those shown in UFSAR Figure 9.1-1. A minimum edge-to-edge spacing between fuel assemblies is maintained in adjacent rows. This spacing is the minimum value after allowances are made for rack fabrication tolerances and the predicted deflections resulting from postulated accident conditions.

The stainless steel construction of the storage racks is compatible with the water and the zirconium-clad fuel. The top structure of the racks is designed such that there is no opening between adjacent fuel cavities that is as large as the cross-section of the fuel bundle. In addition, the outer structure of the racks precludes the inadvertent placement of a bundle against the rack closer than the prescribed edge-to-edge spacing.

Two concrete storage cavities are utilized for NFS. Each cavity is approximately 8 feet by 23 feet and contains 45 fuel assemblies in stainless steel racks. Three racks are installed in each cavity, forming a 3x15 array of fuel assemblies.

The rack structure provides at least 10 inches between the top of the active fuel and the top of the rack to preclude criticality in case a fuel assembly is dropped into a horizontal position on the top of the rack. The NFS racks and facilities are qualified as Seismic Category I and will survive a safe shutdown earthquake without loss of safety function.

The following postulated accidents were considered in the design of the NFS racks:

- Flooding - complete immersion of the entire storage array in pure, unborated, room temperature water
- Envelopment of the entire array in a uniform density aqueous foam or mist of optimum density that maximizes the reactivity of the finite array (a condition that could result from firefighting)
- A fuel assembly dropped from a height of 4.5 ft onto the rack that falls horizontally across the top of the rack
- Tensile load of 5000 lbs on the rack

Although the above accident conditions have been postulated, the FHE, NFS racks, and the building arrangement are designed to minimize the possibility of these accidents and the effects resulting from these accidents.

Intermediate Fuel Storage Rack

The IFSR is a four-cavity fuel storage rack in a 1x4 array designed as an intermediate storage location for fuel bundles during refueling. The rack is located in the containment adjacent to the core support barrel laydown area, which provides access to the refueling machine for insertion and removal of fuel bundles.

Each cavity in the IFSR is a stainless steel can 8.69 inches on a side. The cavities are separated by a fuel center-to-center pitch of 18.56 inches. Each of the cavities is open at the bottom to provide thermal cooling for the worst case fuel bundle. The rack structure is designed to maintain $k_{eff} \leq 0.95$ by assuring under all normal and accident conditions, which includes SSE, that the minimum edge distance is not violated and that a fuel bundle cannot violate the 12-inch minimum stand-off distance around the cavities.

New fuel may be stored in the IFSR before being moved into the core. Partially spent fuel may be moved out of the core and stored temporarily in the IFSR to provide spaces for fuel shuffling. Spent fuel may be stored in the IFSR before being sent to the SFP.

Fuel Upender and Transfer Machine

The transfer machine, or carriage, conveys the fuel assemblies through the transfer tube. Two fuel assembly cavities are provided in the fuel carriage to reduce overall fuel handling time. After the refueling machine deposits a spent fuel bundle in the open cavity, it only has to move approximately one foot to pick up the new fuel assembly, which was brought from the fuel building in the other cavity. The handling operation in the fuel building is similar. The dual cavity

arrangement permits both fuel handling machines to travel fully loaded at all times. Fuel assemblies are placed on the transfer carriage in a vertical position, lowered to the horizontal position, moved through the fuel transfer tube on the transfer carriage, and then restored to the vertical position. Wheels support the carriage and allow it to roll on tracks within the transfer tube. The track sections at both ends of the transfer tube are mounted on the upending machines to permit the carriage to be properly positioned at the limits of its travel.

An upending machine is provided at each end of the transfer tube. Each machine consists of a structural support base from which is pivoted an upending straddle frame that engages the two-cavity fuel carrier. Hydraulic cylinders attached to the upending frame that engages the support base rotate the fuel carrier between the vertical and horizontal position. A third fuel assembly was modeled five inches from the transfer carriage to allow for the presence of an additional fuel assembly no closer than five inches from the carriage. (Appendix B of Attachment 8)

New Fuel Elevator

The new fuel elevator is utilized to lower new fuel from the operating floor to the bottom of the pool where it is grappled by the spent fuel handling tool. The elevator is powered by a cable winch and fuel is contained in a simple support structure whose wheels are captured in two rails.

3.2.2 New Fuel Criticality Safety Analysis

Acceptance Criteria

The objective of the criticality safety analysis is to ensure that the fuel storage operations are within the bounds 10 CFR 50.68(b)(2) and 50.68(b)(3):

- The estimated ratio of neutron production to neutron absorption and leakage (k_{eff}) of the fresh fuel in the storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.
- If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k_{eff} corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

Design Approach

Compliance is shown for the NFS racks, IFSR, and FHE by demonstrating that the system k_{eff} does not exceed 0.95 at a 95 percent probability with a 95 percent confidence level. A conservative combination of best estimate and bounding values has been selected to model the fuel in the analysis to ensure that fuel represented by the proposed TS is less reactive than the fuel modeled in the analysis.

Computer Codes

The analysis methodology employs SCALE 6.1.2 (Reference 6.3) with the 238-group cross-section library based on ENDF/B-VII. All analyses performed used the "Fresh fuel without Absorber" validation suite. KENO-Va is used to determine the absolute reactivity of fresh fuel assemblies in the NFS.

The validation of the ENDF/B-VII library with the SCALE 6.1.2 CSAS5 module is documented in Appendix A of Attachment 8. The code validation shows that SCALE 6.1.2 is an accurate tool for calculation of k_{eff} for the applications in this LAR. The benchmark calculations utilize the same computer platform and cross-section libraries that are used for the design basis calculations.

Limiting Fuel Design Selection

There are four potentially limiting fuel designs that have been used at PVNGS. The VAP design is currently in use on site and NGF is planned for use in the future. Therefore, both VAP and NGF designs are considered the two potentially limiting fuel designs. The STD and ANP fuel designs do not need to be addressed for these calculations because they are bounded by the VAP fuel design. Additional information is provided in Section B.4.2 of Attachment 8.

Treatment of Concrete

Concrete is a material which has a large variety of different potential compositions, all of which can be labeled as "concrete." Attachment 8 references the most limiting design for each situation. The analysis has used a bounding treatment for concrete and the methodology remains conservative throughout the life of the concrete. Additional information is provided in Section B.4.3 of Attachment 8.

Biases and Uncertainties

Reactivity biases are known variations between the real and analyzed system and their reactivity impact is added directly to the calculated k_{eff}. Uncertainties are random dispersions around a nominal, measured quantity. Their impact is added to the calculated k_{eff} as the square root of the sum of the squares of the uncertainties. The following biases and uncertainties are accounted for in the analysis. A detailed discussion of biases and uncertainties is provided in Section B.4.5 of Attachment 8.

- Reactivity effect of manufacturing tolerances
- Structural material presence
- Eccentric fuel assembly positioning
- Uncertainty in the predictive capability of SCALE 6.1.2 and the associated cross-section library
- Temperature bias for operating temperature range
- Planar enrichment bias

New Fuel Storage Rack Criticality Safety Analysis

The criticality safety analysis for the NFS rack consists of determining the limiting fuel design under both the fully flooded and optimum moderation condition. Biases and uncertainties for both fully flooded and optimum moderation conditions are calculated using the limiting fuel design. The best estimate k_{eff} of the NFS rack under both full density water and optimum moderation conditions is less than the target k_{eff}. This demonstrates that the NFS rack complies with the requirements of 10 CFR 50.68. Additional information is provided in Section B.4.6 of Attachment 8.

The analysis of the NFS rack has demonstrated that it can be operated in its design capacity without risk of exceeding the maximum reactivity imposed by regulation. The analysis supports use of these components up to a maximum radially averaged enrichment of 4.65 weight percent U-235. All fuel used to date at PVNGS has an initial enrichment of < 4.55 weight percent U-235.

Intermediate Fuel Storage Rack Criticality Safety Analysis

The criticality safety analysis for the IFSR consists of determining the target k_{eff} for the IFSR, then confirming that the best estimate system k_{eff} (plus 2 σ) is below the target k_{eff} with the limiting fuel design. The analysis uses the NGF design because the NGF is more reactive with full density water than VAP fuel. The biases and uncertainties are also calculated using the NGF design. The best estimate k_{eff} of the IFSR is less than the target k_{eff}, which demonstrates that the IFSR complies with the requirements of 10 CFR 50.68. Additional information is provided in Section B.4.7 of Attachment 8.

Fuel Upender, Transfer Machine, and New Fuel Elevator Criticality Safety Analysis

The criticality safety analysis for the fuel upender and transfer machine, and the analysis for the new fuel elevator, demonstrate that they can be used with fresh 4.65 weight percent U-235 fuel without exceeding a k_{eff} of 0.95 at a 95 percent probability, 95 percent confidence level. The design basis fuel is the NGF design. The best estimate k_{eff} of the fuel upender and transfer machine, and the new fuel elevator, is less than the target k_{eff}, which demonstrates compliance with the requirements of 10 CFR 50.68. Additional information is provided in Section B.4.8 of Attachment 8.

3.3 Spent Fuel Pool Transition Plan

The SFP transition will be conducted over a total lapsed time of approximately 24 months with a schedule based on the unit refueling outages. Therefore, APS will insert two sets of TS and TS Bases pages during implementation. One set will be labeled "Before SFP transition" and the other set will be labeled "After SFP transition."

The Spent Fuel Pool Transition Plan is based on a SFP module-by-module transition scheme. Each SFP module that has not been transitioned is governed by the "Before SFP transition" pages. As each module in a SFP is transitioned to the new configuration, the Shift Manager will make an entry in the control room log and declare that module as "transitioned." That particular module is then governed by the "After SFP transition" pages. When all three units have been transitioned, APS will submit an administrative TS change to remove the "before" and "after" pages, and insert the final pages.

APS will transition to the proposed TS 3.7.17 in each of the three units in the following manner:

1. Move fuel assemblies as needed in order to neutronically decouple one module from the balance of the SFP. Analysis demonstrates that one row of empty cells is enough to decouple modules in the SFP.
2. Perform a shuffle of the decoupled module, including installation of NETCO-SNAP-IN® rack inserts.
    a. Some modules may need to be completely emptied of fuel assemblies. The assemblies that must be moved may be stored in the appropriate regions of the rest of the SFP.
    b. Fuel assemblies that already meet the new TS 3.17.17 requirements may not need to be shuffled.
3. Upon completion of the fuel shuffle and NETCO-SNAP-IN® rack insert installation, the Shift Manager will declare the module has transitioned to the new TS 3.7.17 and enter this information into the control room log.
4. Perform additional fuel shuffles, as needed, in order to move fuel from other parts of the SFP to the recently transitioned module.
5. Repeat steps 1 through 4 for all 17 modules.
6. Perform a 100 percent pool verification to confirm the following
    a. Fuel has been properly moved as confirmed by a 100 percent serial number check
    b. Stainless steel L-Inserts are in the locations assumed in the analysis of record
    c. NETCO-SNAP-IN® rack inserts are in their assumed locations
    d. Blocking devices are in their assumed locations

A generic Westinghouse study (Reference 6.6) investigated the adequacy of assuming a distance of one cell pitch for neutronic isolation. The study included NGF, which was determined to be the limiting fuel type used in the PVNGS SFP criticality safety analysis. The results of the study concluded that a distance of approximately 10 cm (3.94 inches) of water was adequate for neutronic decoupling. Given the required separation distance for neutronic decoupling, fuel assemblies separated by a single cell pitch or more at PVNGS are neutronically decoupled.

Once the Spent Fuel Pool Transition Plan has been started in a particular unit, the insert installation and SFP transition shall be executed in a deliberate, safe, and controlled manner until complete in that unit. The transition to the new SFP configuration will be completed in all three units in accordance with the Spent Fuel Pool Transition Plan within two years of the NRC approval date of the amendment or by December 31, 2019, whichever is later. A regulatory commitment regarding transition plan implementation is provided in Attachment 4.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

The regulations in 10 CFR 50.36(c)(2)(ii)(B), *Limiting conditions for operation*, state:

> *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Technical Specification (TS) 3.7.17 currently meets this requirement and will continue to meet this requirement after the proposed changes are approved and implemented.

The regulations in 10 CFR 50.36(c)(4), *Design features*, state:

> 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 described in paragraphs (c)(1), (2), and (3) of this section.

TS 4.3.1 currently meets this requirement and will continue to meet this requirement after the proposed changes are approved and implemented.

The regulations in 10 CFR 50.68, *Criticality accident requirements*, specifically 10 CFR 50.68(b)(1) state:

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

This requirement is currently met by existing PVNGS fuel handling procedures and will continue to be met by the same procedures after the proposed changes are approved and implemented.

The regulations in 10 CFR 50.68(b)(2) state:

> The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

The regulations in 10 CFR 50.68(b)(3) state:

> If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

The regulations in 10 CFR 50.68(b)(4) state:

> If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent

> fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The requirements in 10 CFR 50.68(b) cited above are met by the nuclear criticality safety analyses provided in WCAP-18030-P. The results of the criticality analysis form the basis of the proposed TS 3.7.17 changes. TS 3.7.17 currently meets these requirements and will continue to meet these requirements after the proposed changes are approved and implemented.

The regulations in 10 CFR 50.68(b)(7) state:

> The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

TS 4.3.1.2 currently meets this requirement and will continue to meet this requirement after the proposed changes are approved and implemented.

The regulations in 10 CFR Part 50, Appendix A, *General Design Criteria for Nuclear Power Plants*, Criterion 62, *Prevention of criticality in fuel storage and handling*, state:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

TS 3.7.17 currently meets this requirement and will continue to meet this requirement after the proposed changes are approved and implemented.

The guidance in DSS-ISG-2010-01 is to be used by NRC staff to review nuclear criticality safety analyses for the storage of new and spent nuclear fuel as they apply to applications for license amendments submitted after September 29, 2011.

This license amendment request and WCAP-18030-P were developed using the guidance of DSS-ISG-2010-01.

4.2 Precedent

The analysis methodology for the site-specific criticality analysis employs the PARAGON code, which is approved for use by the NRC (Reference 6.4).

4.3 Significant Hazards Consideration

As required by 10 CFR 50.91(a), *Notice for Public Comment*, an analysis of the issue of no significant hazards consideration using the standards in 10 CFR 50.92, *Issuance of Amendment*, is presented below:

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

Response: No.

The proposed amendment would modify the Palo Verde Nuclear Generating Station (PVNGS) Technical Specifications (TS) to incorporate the results of an updated criticality safety analysis for both new fuel and spent fuel storage. The revised criticality safety analysis provides an updated methodology that allows credit for neutron absorbing NETCO-SNAP-IN® rack inserts and corrects non-conservative input assumptions in the previous criticality safety analysis.

The proposed amendment does not change or modify the fuel, fuel handling processes, number of fuel assemblies that may be stored in the spent fuel pool (SFP), decay heat generation rate, or the SFP cooling and cleanup system. The proposed amendment was evaluated for impact on the following previously evaluated events and accidents:

- fuel handling accident (FHA)
- fuel misload event
- SFP boron dilution event
- seismic event
- loss of SFP cooling event

Implementation of the proposed amendment will be accomplished in accordance with the Spent Fuel Pool Transition Plan and does not involve new fuel handling equipment or processes. The radiological source term of the fuel assemblies is not affected by the proposed amendment request. The FHA radiological dose consequences associated with fuel enrichment at this level are addressed in the PVNGS Updated Final Safety Analysis Report (UFSAR) Section 15.7.4 and remain unchanged. Therefore, the proposed amendments do not significantly increase the probability or consequences of a FHA.

Operation in accordance with the proposed amendment will not change the probability of a fuel misload event because fuel movement will continue to be controlled by approved fuel handling procedures. Although there will be additional allowable storage arrays defined by the amendment, the fuel handling procedures will continue to require identification of the initial and target locations for each fuel assembly that is moved. The consequences of a fuel misload event are not changed because the reactivity analysis demonstrates that the same subcriticality criteria and requirements continue to be met for the limiting fuel misload event.

Operation in accordance with the proposed amendment will not change the probability or consequences of a boron dilution event because the systems and events that could affect SFP soluble boron concentration are unchanged. The current boron dilution analysis demonstrates that the limiting boron dilution event will reduce the boron concentration from the TS limit of 2150 ppm to 1900 ppm. This leaves sufficient margin to the 1460 ppm credited by the SFP criticality safety analysis. The analysis confirms that the time needed for dilution to reduce the soluble boron concentration is greater than the time needed for actions to be taken to prevent further dilution.

Operation in accordance with the proposed amendment will not change the probability of a seismic event since there are no elements of the updated criticality analysis that influence the occurrence of a seismic event. The consequences of a seismic event are not significantly increased because the forcing functions for seismic excitation are not increased and because the mass of storage racks with NETCO-SNAP-IN® inserts is not appreciably

increased. Seismic analyses demonstrate adequate stress levels in the storage racks when inserts are installed.

Operation in accordance with the proposed amendment will not change the probability of a loss of SFP cooling event because the systems and events that could affect SFP cooling are unchanged. The consequences are not significantly increased because there are no changes in the SFP heat load or SFP cooling systems, structures, or components. Furthermore, conservative analyses indicate that the current design requirements and criteria continue to be met with the NETCO-SNAP-IN® inserts installed.

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

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

Response: No.

The proposed amendment would modify the PVNGS TS to incorporate the results of an updated criticality safety analysis for both new fuel and spent fuel storage. The revised criticality safety analysis provides an updated methodology that allows credit for neutron absorbing NETCO-SNAP-IN® rack inserts and corrects non-conservative input assumptions in the previous criticality safety analysis.

The proposed amendment does not change or modify the fuel, fuel handling processes, number of fuel assemblies that may be stored in the pool, decay heat generation rate, or the SFP cooling and cleanup system. The effects of operating with the proposed amendment are listed below. The proposed amendment was evaluated for the potential of each effect to create the possibility of a new or different kind of accident:

- addition of inserts to the SFP storage racks
- additional weight from the inserts
- new storage patterns
- displacement of SFP water by the inserts,

Each NETCO-SNAP-IN® insert will be placed between a fuel assembly and the storage cell wall, taking up some of the space available on two sides of the fuel assembly. Analyses demonstrate that the presence of the inserts does not adversely affect spent fuel cooling, seismic capability, or subcriticality. The aluminum and boron carbide materials of construction have been shown to be compatible with nuclear fuel, storage racks, and SFP environments, and generate no adverse material interactions. Therefore, placing the inserts into the SFP storage racks cannot cause a new or different kind of accident.

Operation with the added weight of the NETCO-SNAP-IN® inserts will not create a new or different accident. The analyses of the racks with NETCO-SNAP-IN® inserts installed demonstrate that the stress levels in the rack modules continue to be considerably less than allowable stress limits. Therefore, the added weight from the inserts cannot cause a new or different kind of accident.

Operation with the proposed fuel storage patterns will not create a new or different kind of accident because fuel movement will continue to be controlled by approved fuel handling procedures. These procedures continue to require identification of the initial and target locations for each fuel assembly that is moved. There are no changes in the criteria or design requirements pertaining to fuel storage safety, including subcriticality requirements. Analyses demonstrate that the proposed storage patterns meet these requirements and criteria with adequate margins. Therefore, the proposed storage patterns cannot cause a new or different kind of accident.

Operation with insert movement above stored fuel will not create a new or different kind of accident. The insert with its handling tool weighs less than the weight of a single fuel assembly. Single fuel assemblies are routinely moved safely over fuel assemblies and the same level of safety in design and operation will be maintained when moving the inserts. The installed rack inserts will displace a negligible quantity of the SFP water volume and therefore will not reduce operator response time to previously-evaluated SFP accidents.

The accidents and events previously analyzed remain bounding. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed amendment would modify the TS to incorporate the results of an updated criticality safety analysis for both new fuel and spent fuel storage. The revised criticality safety analysis provides an updated methodology that allows credit for neutron absorbing NETCO-SNAP-IN® rack inserts and corrects non-conservative input assumptions in the previous criticality safety analysis. It was evaluated for its effect on current margins of safety as they relate to criticality, structural integrity, and spent fuel heat removal capability. The margin of safety for subcriticality required by 10 CFR 50.68(b)(4) is unchanged. New criticality analyses confirm that operation in accordance with the proposed amendment continues to meet the required subcriticality margins.

The structural evaluations for the racks and spent fuel pool with NETCO-SNAP-IN® inserts installed show that the rack and SFP are unimpaired by loading combinations during seismic motion, and there is no adverse seismic-induced interaction between the rack and NETCO-SNAP-IN® inserts.

The proposed amendment does not affect spent fuel heat generation, heat removal from the fuel assembly, or the SFP cooling systems. The effects of the NETCO-SNAP-IN® inserts are negligible with regards to volume of water in the pool, flow in the SFP rack cells, and heat removal system performance.

The addition of a Spent Fuel Pool Rack Neutron Absorber Monitoring program (proposed TS 5.5.21) provides a method to identify potential degradation in the neutron absorber material prior to challenging the assumptions of the criticality safety analysis related to the material. Therefore, the addition of this monitoring program does not reduce the margin of safety;

rather it ensures the margin of safety is maintained for the planned life of the spent fuel storage racks.

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

4.4 Conclusion

APS concludes that operation of the facility in accordance with the proposed amendment does not involve a 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. 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 the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would 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*. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a 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 amendment meets 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 amendment.

6.0 REFERENCES

6.1 *Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3 (Proprietary)*, WCAP-18030-P, Revision 0, September 2015.

6.2 *Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,* DSS-ISG-2010-01, Revision 0, Nuclear Regulatory Commission Division of Safety Systems, Rockville, MD, September 29, 2011. (ML110620086)

6.3 *Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,* ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, TN, June 2011.

6.4 M. Ouisloumen, H. Huria, et al, *Qualification of the Two-Dimensional Transport Code PARAGON*, WCAP-16045-P-A, Revision 0, Westinghouse Electric Company LLC, Monroeville, PA, August 2004.

6.5 C. V. Parks, et al, *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel,* NUREG/CR-6665, Oak Ridge National Laboratory, Oak Ridge, TN, February 2000.

6.6 Letter, J. Gresham (WEC) to NRC, *Responses to Requests for Additional Information from the Review of WCAP-17483-P/WCAP-17483-NP, Revision 0, 'Westinghouse Methodology for Spent Fuel Pool and New Fuel Rack Criticality Safety Analysis,'* LTR-NRC-15-60, dated July 20, 2015.

ATTACHMENT 1

Marked-up Technical Specifications Pages

(Pages Provided for Before and After SFP Transition)

3.7.17-1
3.7.17-2
3.7.17-3
3.7.17-4
4.0-2
4.0-3
5.5-19

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.

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

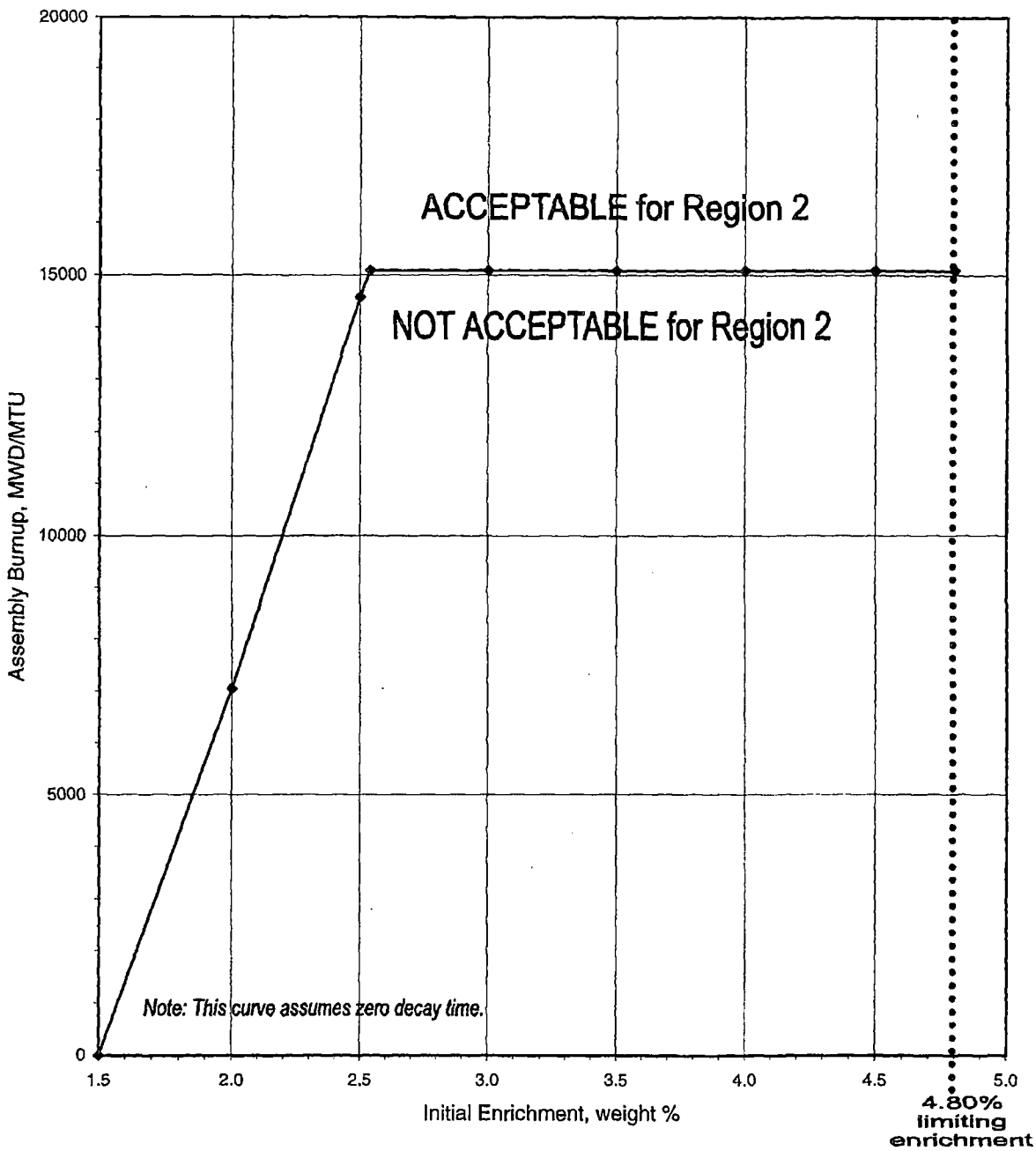


Figure 3.7.17-2
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 3
(at decay times from 0 to 20 years)

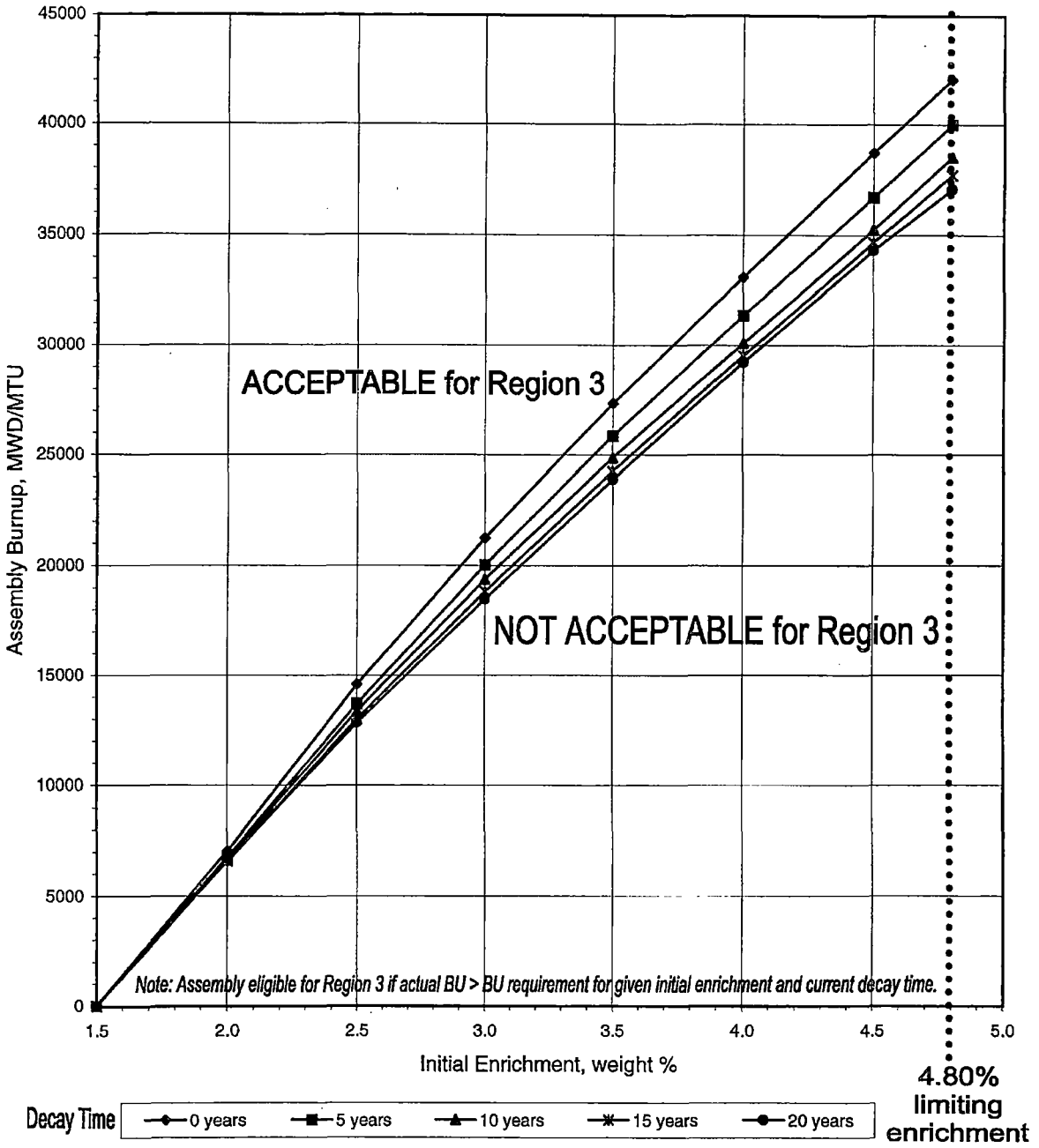
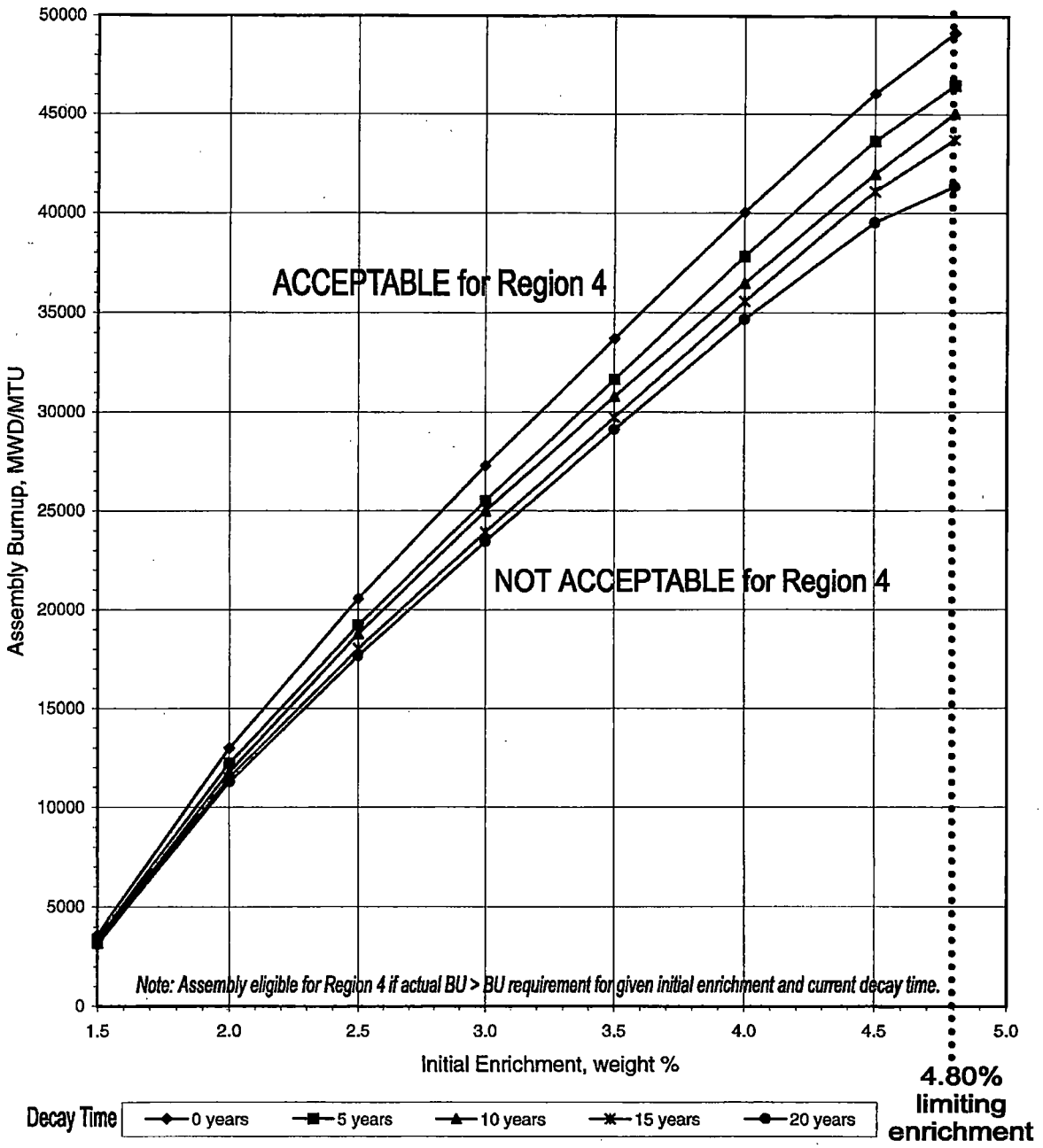


Figure 3.7.17-3
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 4
(at decay times from 0 to 20 years)



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. ~~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

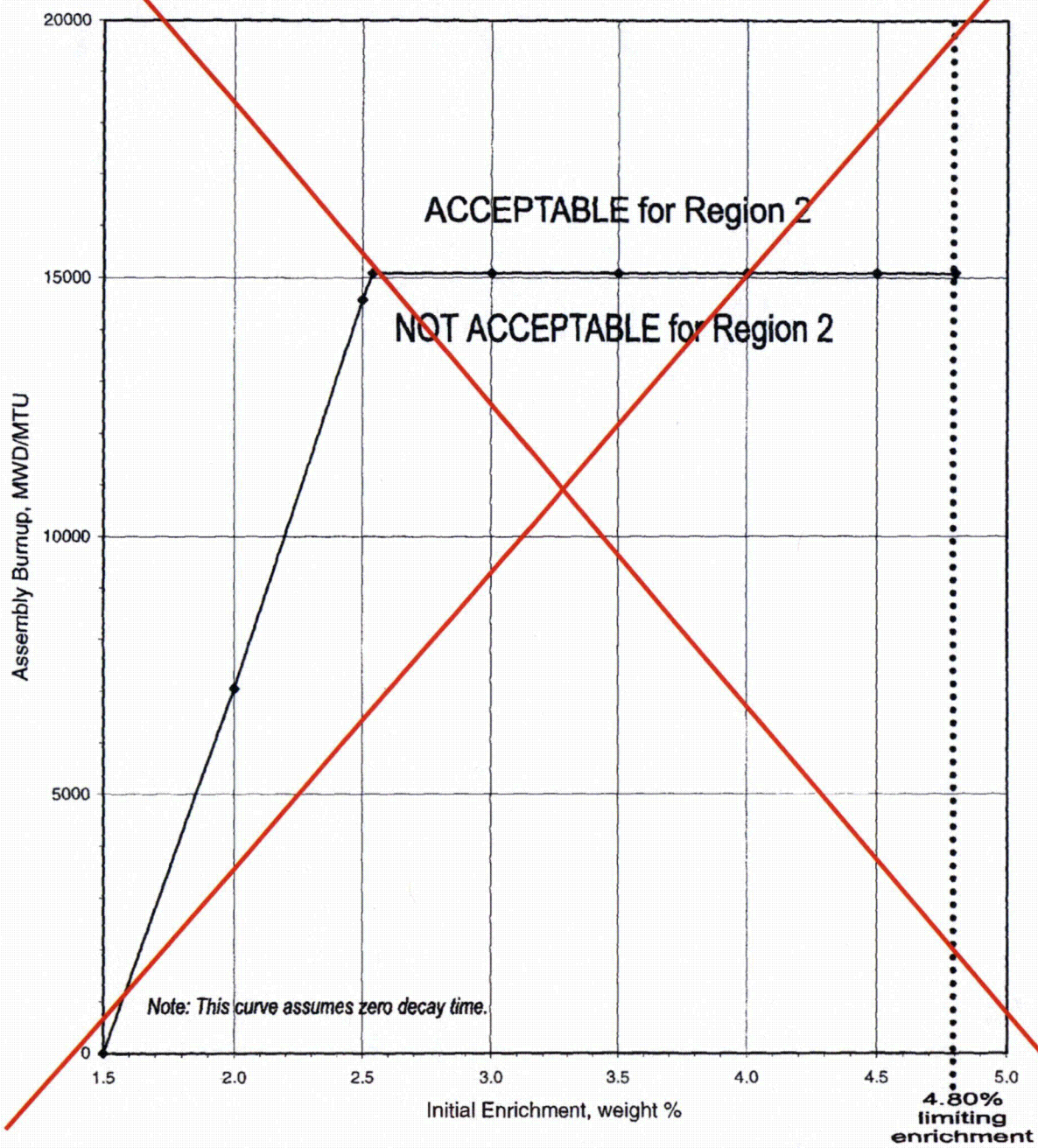
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE--------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly into an appropriate region.	Immediately

SURVEILLANCE REQUIREMENTS

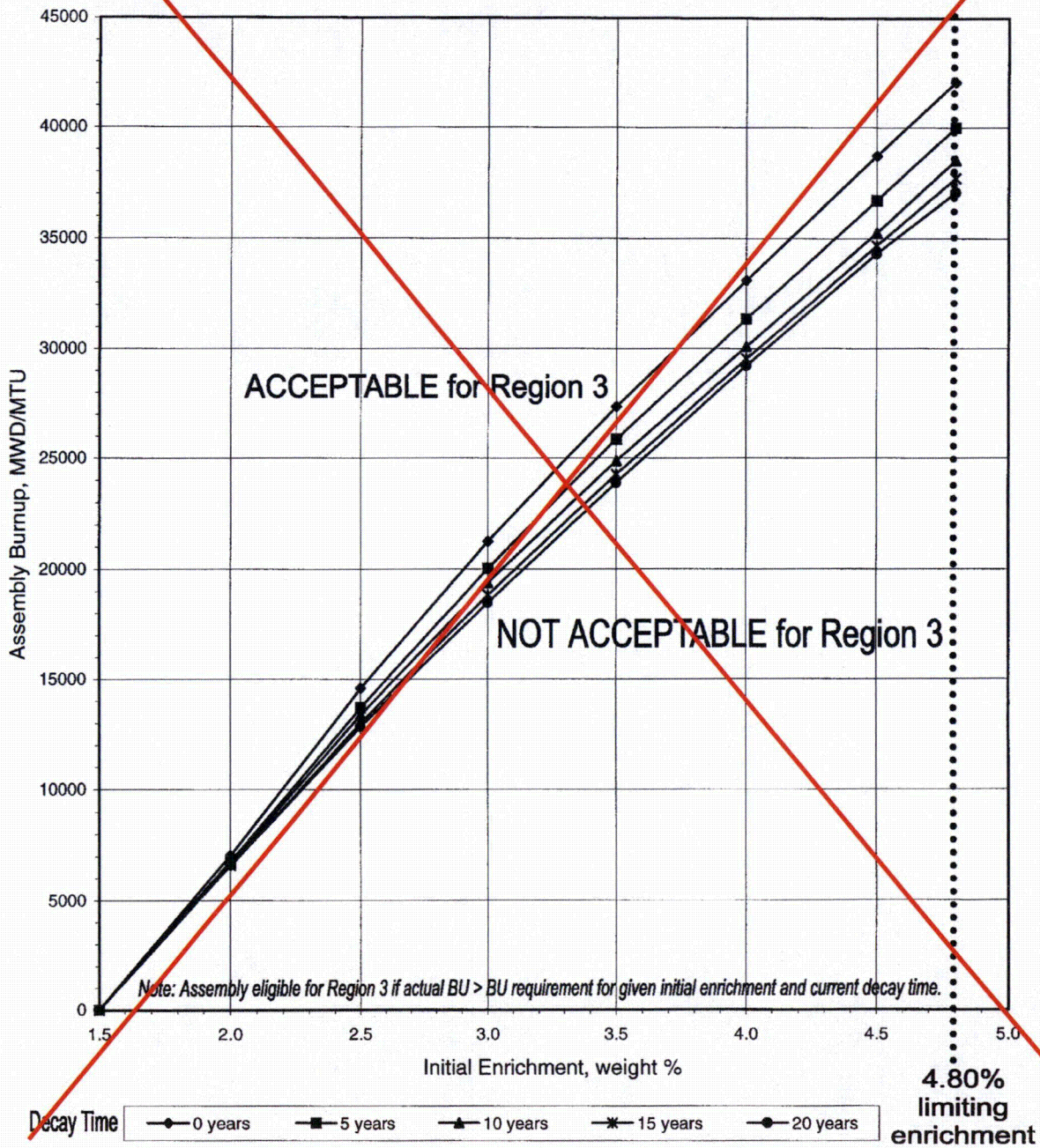
SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with Tables 3.7.17-1 through 3.7.17-5, Figure 3.7.17-1, ~~Figures 3.7.17-1, 3.7.17-2, or 3.7.17-3,~~ and Specification 4.3.1.1.	Prior to storing the fuel assembly in the fuel storage pool.

Insert new Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1 (total 6 pages) here.

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



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



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

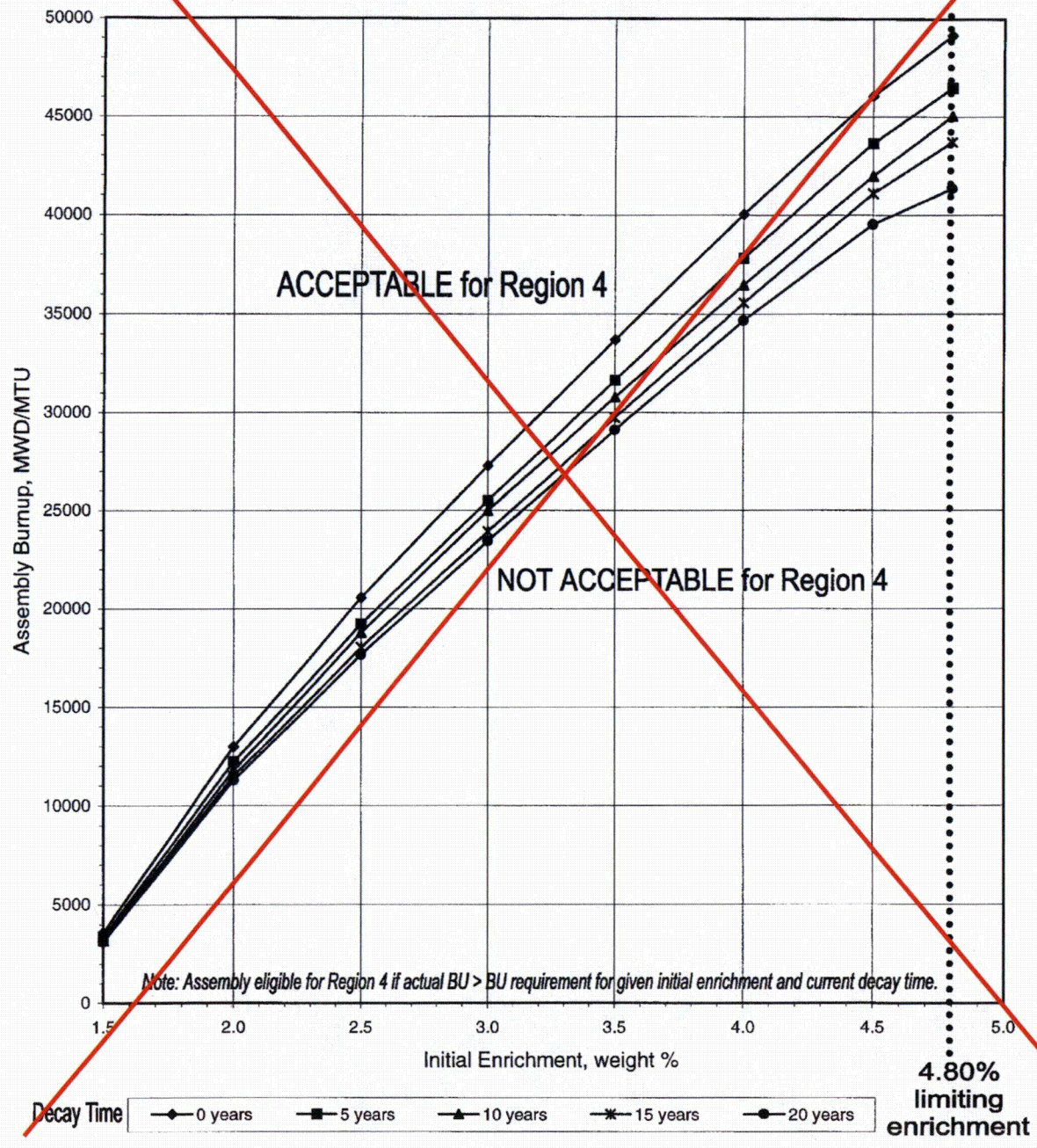


Table 3.7.17-1

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	-1.5473	15.5395	-39.0197	24.1121
5	-1.4149	13.9760	-33.6287	18.3369
10	-1.3012	12.6854	-29.2539	13.6879
15	-1.0850	10.4694	-22.1380	6.3673
20	-0.9568	9.1487	-17.9045	2.0337

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 2.55 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 2.55 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	0.4260	-6.2766	40.9264	-54.6813
5	0.2333	-4.1545	32.9080	-46.1161
10	0.4257	-6.2064	39.0371	-51.5889
15	0.5315	-7.3777	42.5706	-54.7524
20	0.5222	-7.3897	42.6587	-54.8201

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	-0.1114	-0.4230	20.9136	-32.8551
5	-0.1232	-0.4463	20.8337	-32.6068
10	-0.2357	0.4892	18.0192	-30.0042
15	-0.1402	-0.4523	20.3745	-31.7565
20	-0.0999	-0.8152	21.0059	-31.9911

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	0.7732	-9.3583	49.6577	-54.6847
5	0.7117	-8.4920	45.1124	-49.7282
10	0.6002	-7.2638	40.2603	-44.9348
15	0.5027	-6.2842	36.6715	-41.4934
20	0.2483	-3.7639	28.8269	-34.6419

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 and only water in the active fuel region.
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.

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 4.80 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)

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.

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 ~~4.80~~ 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} ≤ 0.95 if fully flooded with water borated to ~~900~~ 1460 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.~~

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

~~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)

~~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~~ 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 ~~17 inch center to center~~ 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 (continued)

5.5.19 Battery Monitoring and Maintenance Program (continued)

4. In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

5.5 Programs and Manuals (continued)

5.5.19 Battery Monitoring and Maintenance Program (continued)

4. In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

Insert for page 5.5-19

Insert for page 5.5-19

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.

ATTACHMENT 2

Revised Technical Specifications Pages (Clean Copy)

(Pages Provided for Before and After SFP Transition)

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
4.0-2
4.0-3
5.5-19
5.5-20

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.

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

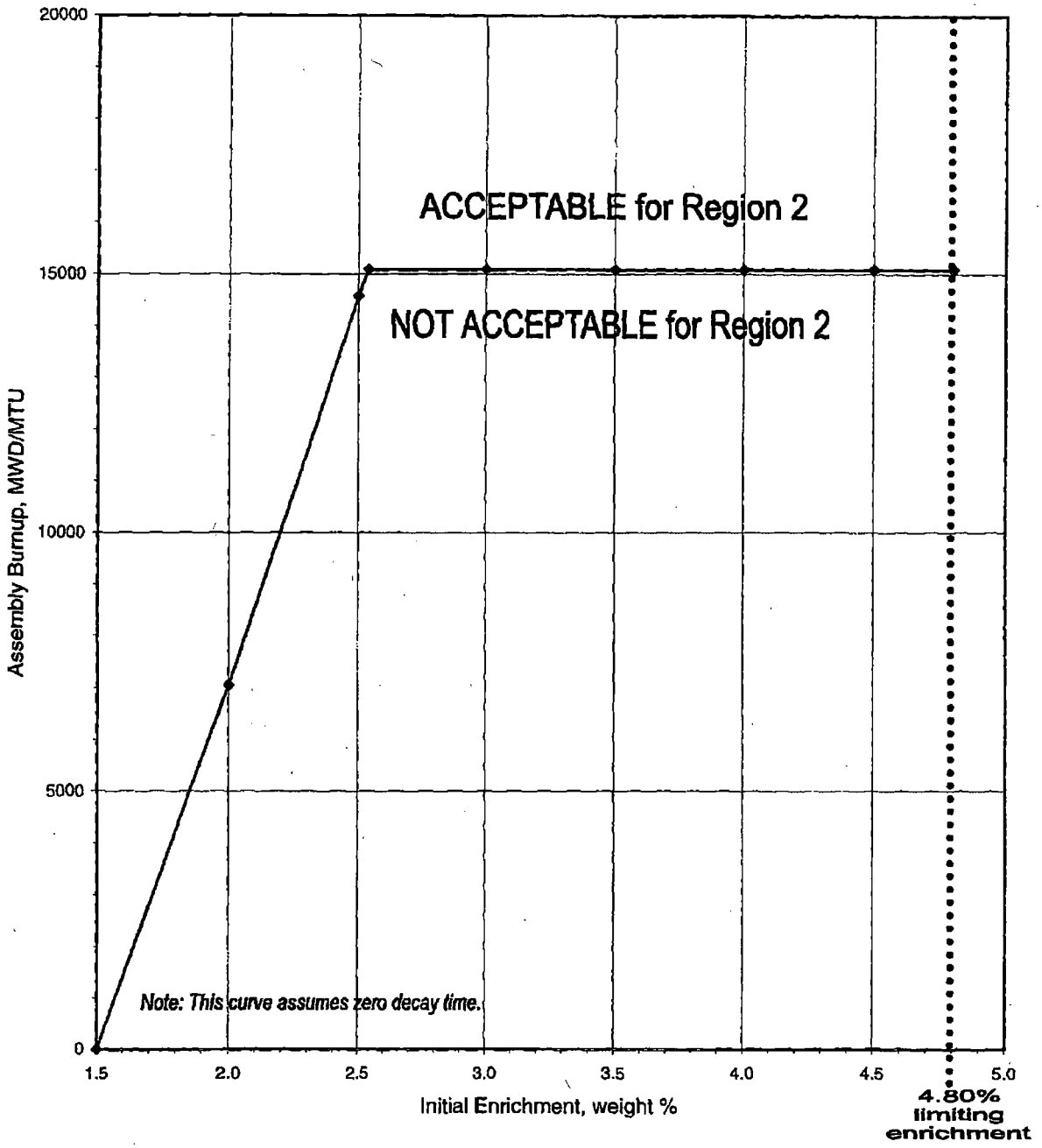


Figure 3.7.17-2
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 3
(at decay times from 0 to 20 years)

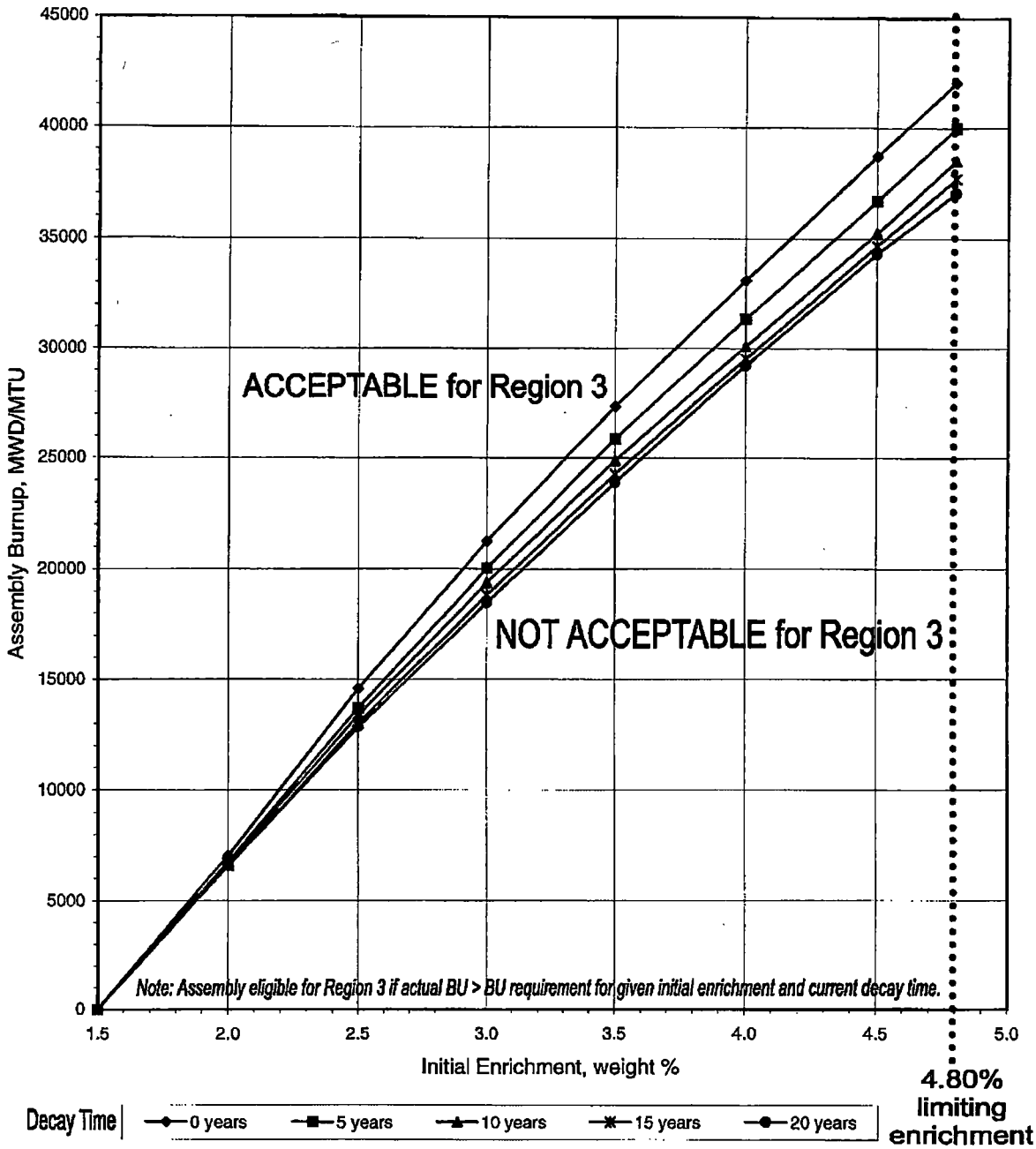
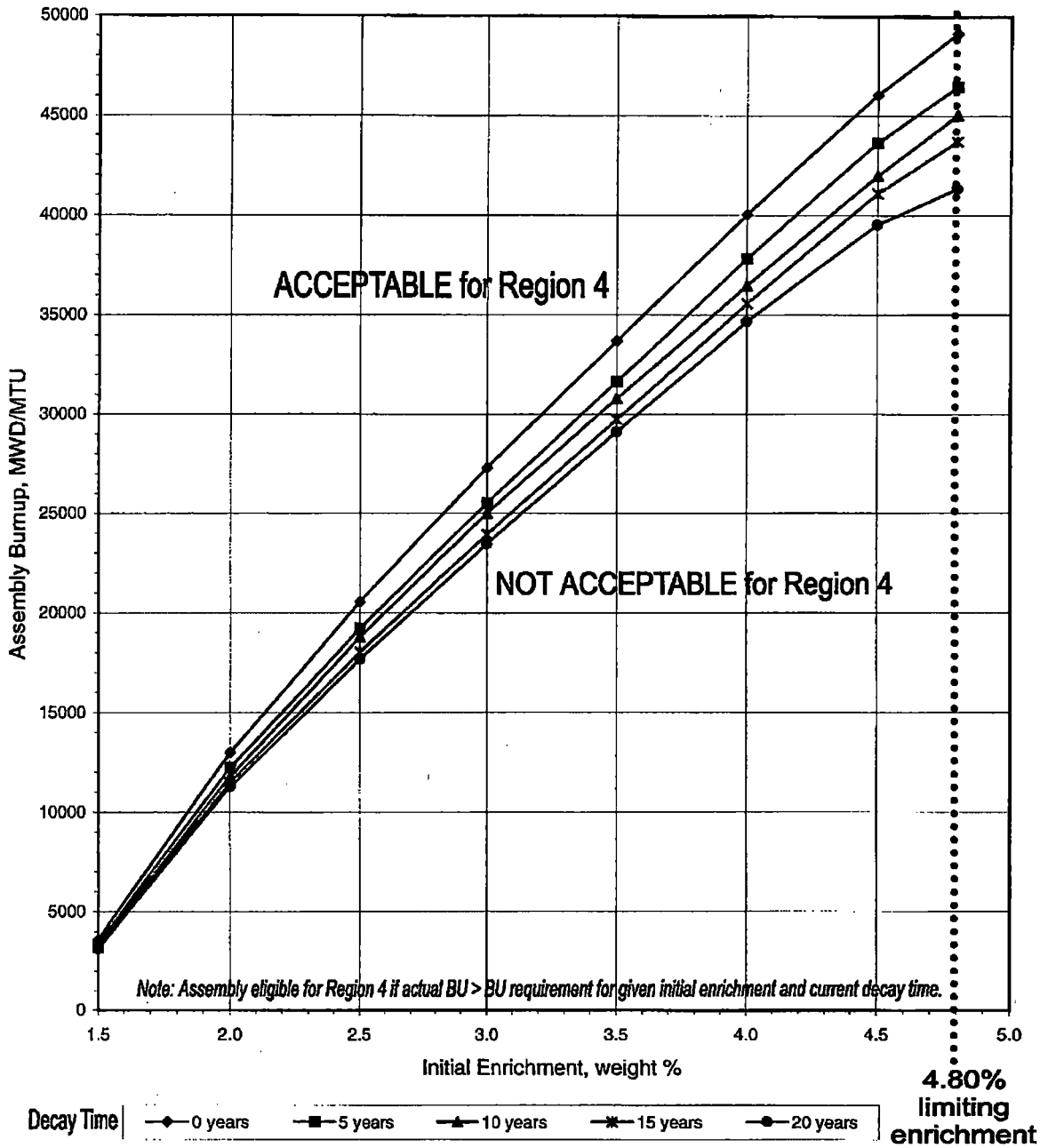


Figure 3.7.17-3
ASSEMBLY BURNUP VERSUS INITIAL ENRICHMENT
for
Region 4
(at decay times from 0 to 20 years)



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	-1.5473	15.5395	-39.0197	24.1121
5	-1.4149	13.9760	-33.6287	18.3369
10	-1.3012	12.6854	-29.2539	13.6879
15	-1.0850	10.4694	-22.1380	6.3673
20	-0.9568	9.1487	-17.9045	2.0337

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 2.55 wt% ^{235}U and 4.65 wt% ^{235}U may be used. Burnup credit is not required for an En below 2.55 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	0.4260	-6.2766	40.9264	-54.6813
5	0.2333	-4.1545	32.9080	-46.1161
10	0.4257	-6.2064	39.0371	-51.5889
15	0.5315	-7.3777	42.5706	-54.7524
20	0.5222	-7.3897	42.6587	-54.8201

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	-0.1114	-0.4230	20.9136	-32.8551
5	-0.1232	-0.4463	20.8337	-32.6068
10	-0.2357	0.4892	18.0192	-30.0042
15	-0.1402	-0.4523	20.3745	-31.7565
20	-0.0999	-0.8152	21.0059	-31.9911

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	0.7732	-9.3583	49.6577	-54.6847
5	0.7117	-8.4920	45.1124	-49.7282
10	0.6002	-7.2638	40.2603	-44.9348
15	0.5027	-6.2842	36.6715	-41.4934
20	0.2483	-3.7639	28.8269	-34.6419

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 and only water in the active fuel region.
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.

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 4.80 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)

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.

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 1460 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 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 (continued)

5.5.19 Battery Monitoring and Maintenance Program (continued)

4. In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

5.5 Programs and Manuals (continued)

5.5.19 Battery Monitoring and Maintenance Program (continued)

4. In Regulatory Guide 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."

5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

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.

ATTACHMENT 3

Marked-up Technical Specifications Bases Pages

(Pages Provided for Before and After SFP Transition)

B 3.7.15-1
B 3.7.15-2
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 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)

BASES (continued)

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

, and decay time.

,

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} ≤ 0.95 assuming the most limiting ~~single~~ fuel mishandling accident.

A

1460

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.

There could also be a misload of multiple fuel assemblies into fuel rack locations not allowed by LCO 3.7.17.

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} ≤ 0.95.

(continued)

BASES (continued)

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.~~

"Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3" (Proprietary), WCAP-18030-P, Revision 0, September 2015.

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. Cell blocking devices are placed in every other storage cell location in Region 1 to maintain a two-out-of-four checkerboard configuration. These cell blocking devices prevent inadvertent insertion of a fuel assembly into a cell that is not allowed to contain a fuel assembly.

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)

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 ≤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 UO(2) density of 10.31 g/cc.

(continued)

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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)

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)

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.

. 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).

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly

BASES

pool

and

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. Cell blocking devices are placed in every other storage cell location in Region 1 to maintain a two-out-of-four checkerboard configuration. These cell blocking devices prevent inadvertent insertion of a fuel assembly into a cell that is not allowed to contain a fuel assembly.~~

~~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.~~

Insert 1

~~(continued)~~

Insert 1 for TS Bases 3.7.17 page B 3.7.17-1

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 1460 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%.

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.

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 ≤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} ≤ 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} ≤ 0.95 assuming a soluble boron concentration of 900 ppm and the most limiting single fuel mishandling accident.~~

Insert 2

~~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 UO(2) density of 10.31 g/cc.~~

~~(continued)~~

Insert 2 for TS Bases 3.7.17 page B 3.7.17-2

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

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.

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

~~(continued)~~

BASES

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~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).

Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1

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~~ six Regions.

(continued)

BASES

APPLICABILITY

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

ACTIONS

A.1

Tables 3.7.17-1 through 3.7.17-5 and Figure 3.7.17-1

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

, discharge burnup, and decay time

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:~~

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.

(continued)

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.~~

"Criticality Safety Analysis for Palo Verde Nuclear Generating Station Units 1, 2, and 3" (Proprietary), WCAP-18030-P, Revision 0, September 2015.

ATTACHMENT 4

List of Regulatory Commitments

List of Regulatory Commitments

Regulatory Commitment	Due Date/Milestone
1. APS will implement procedural controls to require verification that fresh fuel assemblies are not placed face-adjacent to one another before completing a fuel move.	Within the requested 90-day implementation period following NRC approval.
2. The transition to the new SFP configuration will be completed in all three units in accordance with the Spent Fuel Pool Transition Plan within two years of the NRC approval date of the amendment or by December 31, 2019, whichever is later.	Within two years of the NRC approval date of the amendment or by December 31, 2019, whichever is later.