161-04355-WFC/CAM
December 24, 1991

Replace with
revised Figure 3.2-2

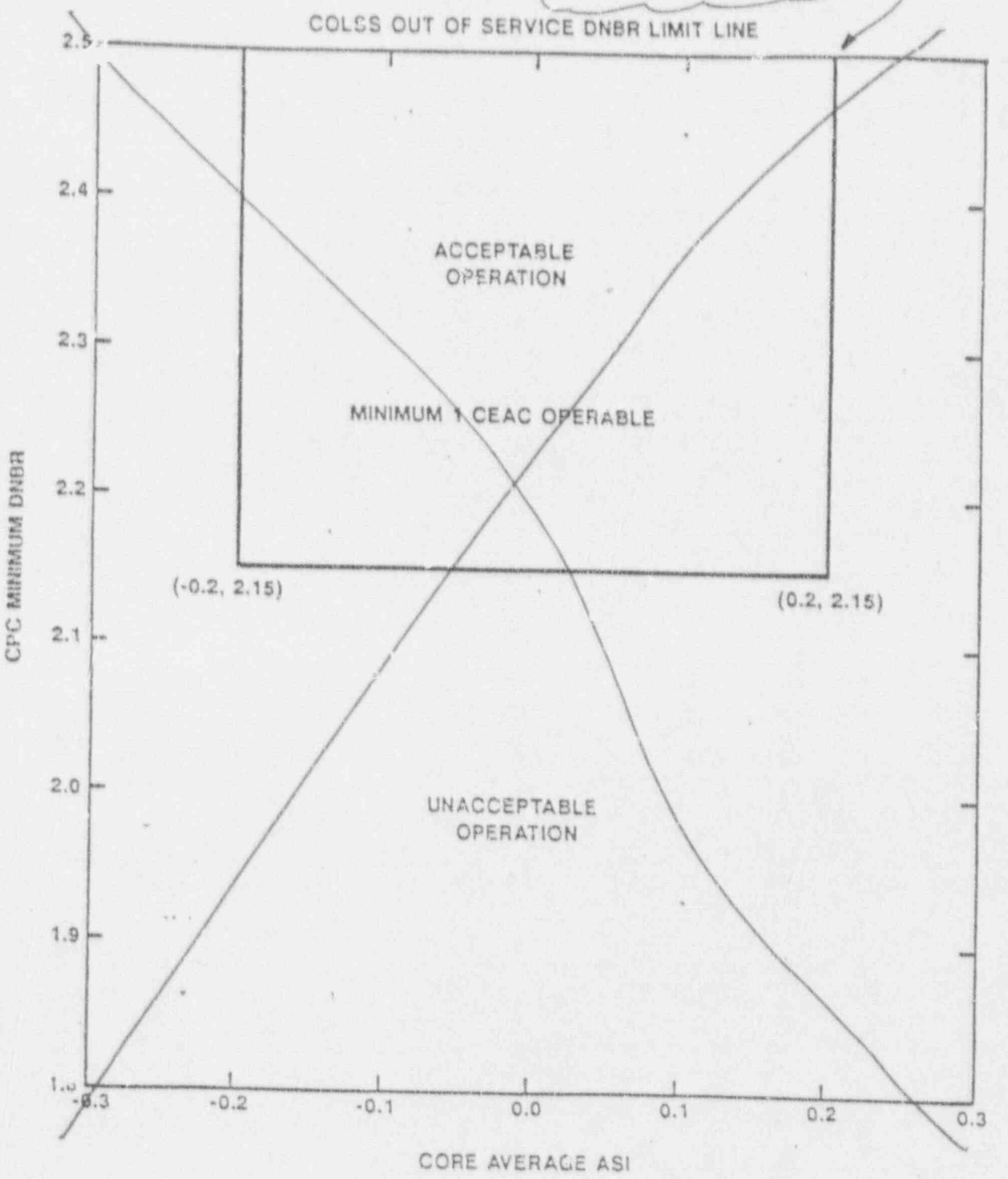

FIGURE 3.2-2

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (COLSS OUT OF SERVICE, CEAC'S OPERABLE)

Revised Figure 3.

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COLSS OUT OF SERVICE DNBR LIMIT LINE

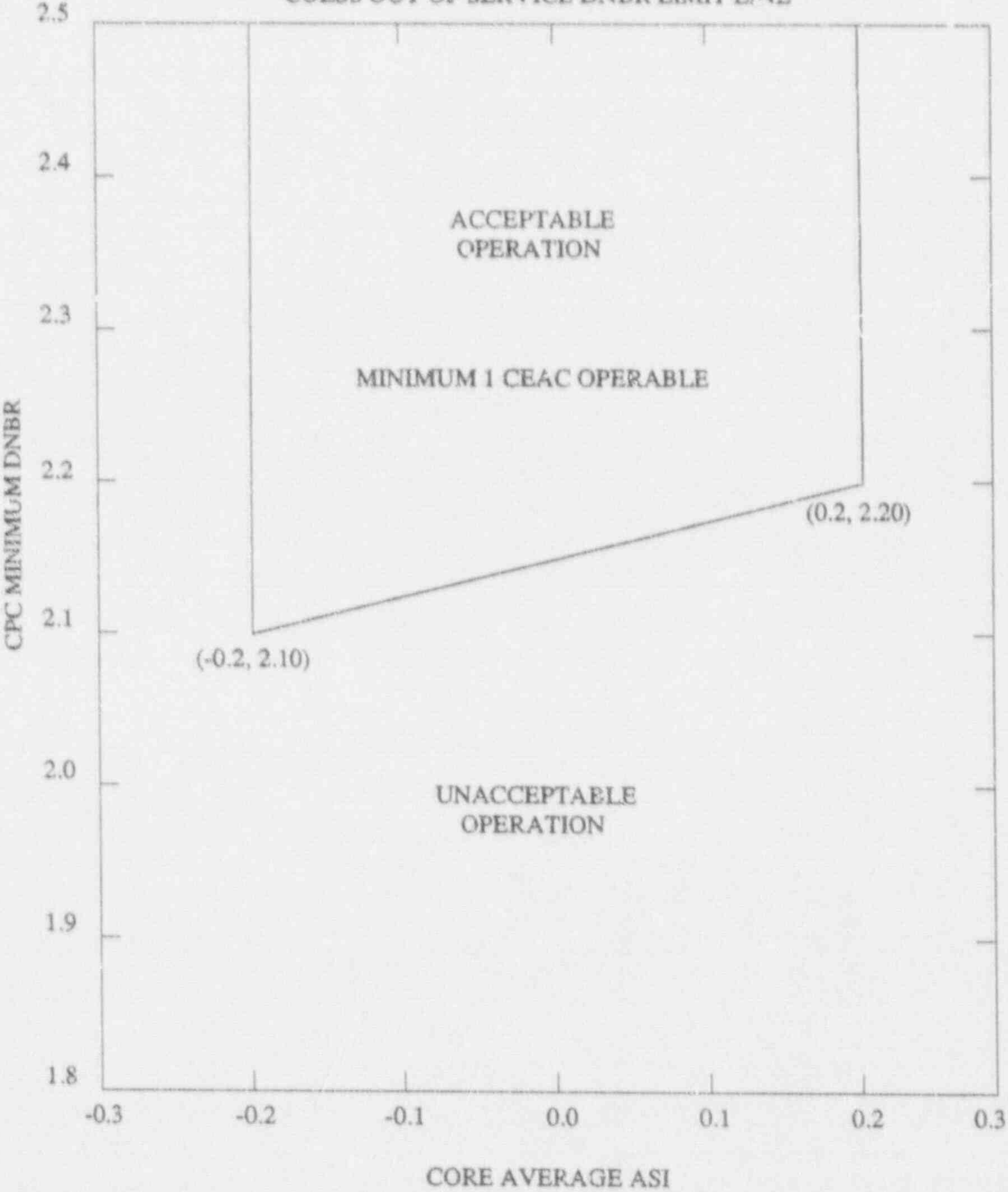

FIGURE 3.2-2
DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEAC's OPERABLE)

Replace with revised Figure 3.2-2a

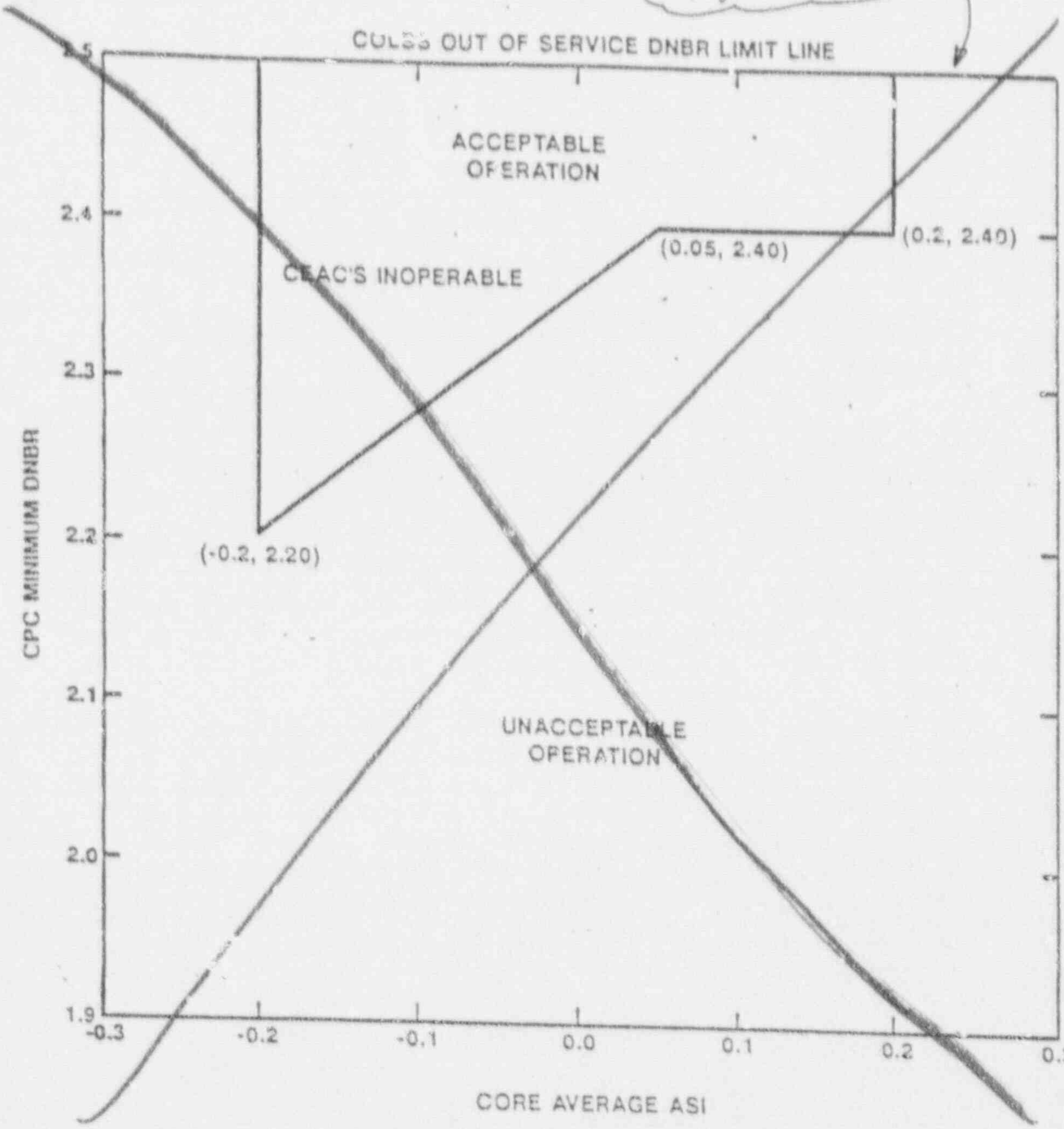

FIGURE 3.2-2a

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (COLSS OUT OF SERVICE, CEAC'S INOPERABLE)

Revised
Figure 3.2-2a

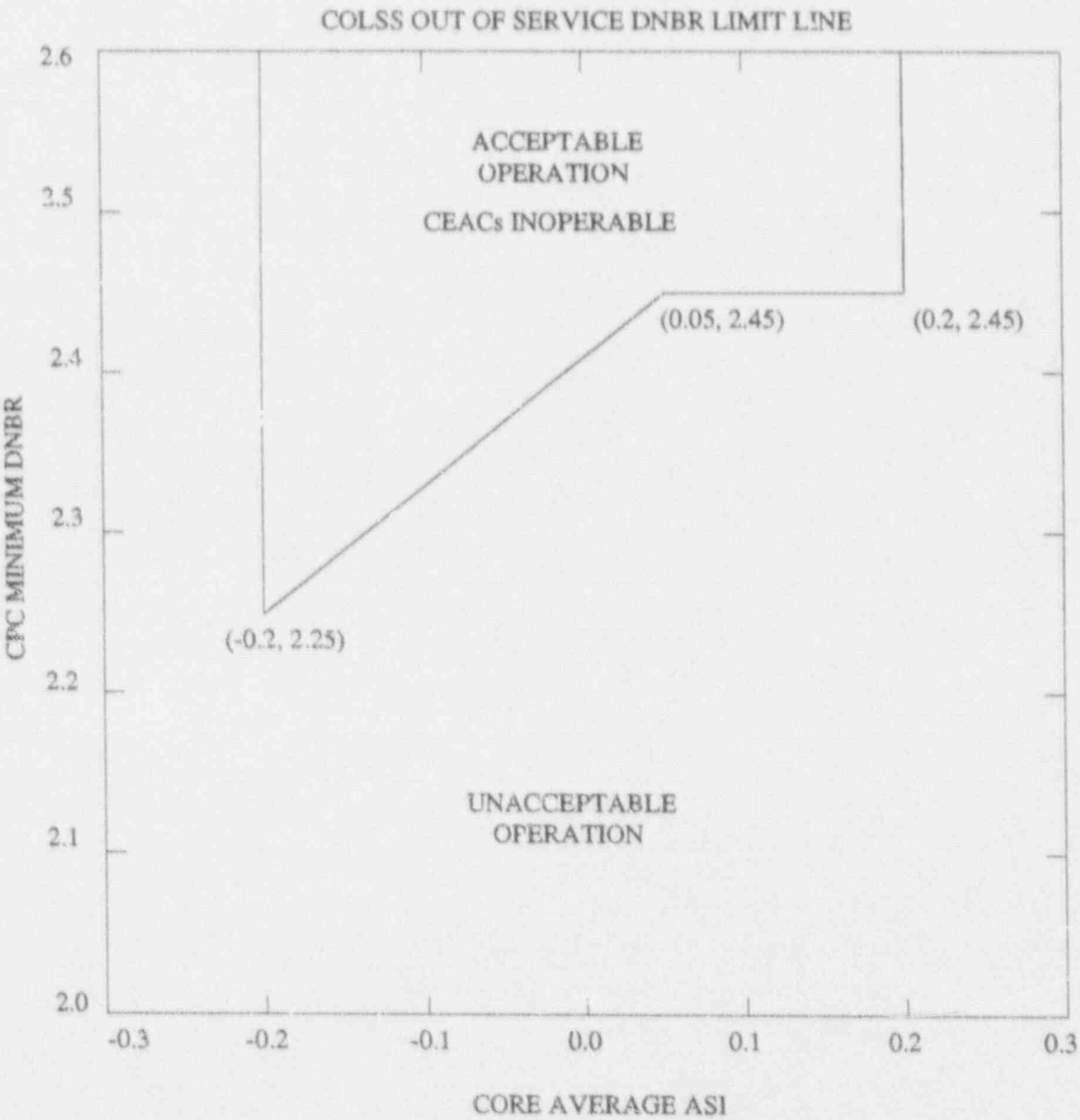

FIGURE 3.2-2a

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEACs INOPERABLE)

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ENCLOSURE B

UNIT 1 CYCLE 4 RELOAD ANALYSIS REPORT

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RELOAD ANALYSIS REPORT FOR PALO VERDE NUCLEAR GENERATING STATION UNIT 1 CYCLE 4

TABLE OF CONTENTS

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance of Palo Verde Nuclear Generating Station Unit 1 (PVNGS-1) during its fourth cycle of operation at 100% rated core power of 3800 MWt and NSSS power of 3822 MWt. Operating conditions for Cycle 4 have been assumed to be consistent with those of the previous cycle and are summarized as full power operation under base load conditions. The core will consist of irradiated Batch B, C, D, and E assemblies, along with fresh Batch F assemblies. The Cycle 3 termination burnup has been assumed to be between 465 and 517 EFPD (Effective Full Power Days).

The third cycle of operation will hereafter be referred to in this report as the "Reference Cycle." Reference 1-2 presented analyses for the Reference Cycle.

The safety criteria (margins of safety, dose limits, etc.) applicable for the plant were established in Reference 1-1. A review of those postulated accidents and anticipated operational occurrences evaluated in Reference 1-1 has shown that the Cycle 4 core design meets these safety criteria.

The Cycle 4 reload core characteristics have been evaluated with respect to the Reference Cycle. Specific differences in core fuel loadings have been accounted for in the present analysis. The status of the postulated accidents and anticipated operational occurrences for Cycle 4 can be summarized as follows:

1. Transient data are less severe than those of the Reference Cycle analysis; therefore, no reanalysis is necessary, or

2. Transient data are <u>not</u> bounded by those of the Reference Cycle analysis, therefore, reanalysis is required.

For those transients requiring reanalysis (Type 2), analyses are presented in Sections 7 and 8 showing results that meet the established safety criteria.

The Technical Specification changes needed for Cycle 4 are summarized in Section 10.

2.0 OPERATING HISTORY OF THE REFERENCE CYCLE

The Reference Cycle began with initial criticality on June 24, 1990. Power Ascension began on June 28, 1990, and on July 12, 1990 the unit reached full power.

It is presently estimated that Cycle 3 will terminate on or about February 1, 1992. The Cycle 3 termination point can vary between 465 and 517 EFPD to accommodate the plant schedule and still be within the assumptions of the Cycle 4 analyses.

3.0 GENERAL DESCRIPTION

The Cycle 4 core will consist of those assembly types and numbers listed in Table 3-1. One Batch B assembly, fifty-two Batch C assemblies, and forty-four Batch D assemblies will be removed from the Cycle 3 core to make way for eighty-eight fresh Batch F assemblies. 108 Batch E and 36 Batch D assemblies now in the core will be retained. In addition, 5 Batch B assemblies originally discharged at EOC1 and 4 Batch C assemblies originally discharged at EOC2 will be reinserted from the spent fuel storage. Figure 3-1 shows the poison shim and zoning configuration for the discharged assemblies.

The reload batch will consist of 4 type F0 assemblies, 24 type F1 assemblies with 4 burnable poison shims per assembly, 4 type F2 assemblies with 12 burnable poison shims per assembly, 8 type F3 assemblies with 8 burnable poison shims per assembly, 16 type F4 assemblies with 16 burnable poison shims per assembly, and 32 type F5 assemblies with 12 burnable poison shims per assembly. These sub-batch types are fuel zone-enriched and their assembly configurations are shown in Figure 3-2.

The loading pattern for Cycle 4, showing fuel type and location, is displayed in Figure 3-3.

Figure 3-4 displays the beginning of Cycle 4 assembly average burnup distribution. The burnup distribution is based on a Cycle 3 length of 517 EFPD, which is the long endpoint of Cycle 3.

Control element assembly patterns and in-core instrument locations will remain unchanged from the Reference Cycle and are shown in Figures 3-5 A & B and Figure 3-6, respectively.

TABLE 3-1

PALO VERDE NUCLEAR GENERATING STATION UNIT 1
CYCLE 4 CORE LOADING

Assembly Desig-nation	Number of Assemblies	[illegible]	Nominal Initi[illegible] [illegible]richm[illegible] [illegible]35)	Number Shims/ Assembly	Shim loading (gm B-10/in)	Number of Fuel Rods	Number of Shim Rods
B	5	[illegible]	[illegible]	16	0.018	1040	80
		[illegible]	[illegible]			60	
C	4	22[illegible]	[illegible]	0	——	896	0
		12	[illegible]			48	
D	36	184	4.05	0	——	6624	0
		52	3.36			1872	
E0	24	184	4.03	0	——	4416	0
		52	3.90			1248	
E1	20	168	4.03	16	0.024	3360	320
		52	3.90			1040	
E2	12	168	3.90	16	0.024	2016	192
		52	3.60			624	
E3	12	168	3.90	16	0.026	2016	192
		52	3.60			624	
E4	24	168	3.90	16	0.016	4032	384
		52	3.60			1248	
E5	8	180	4.03	4	0.012	1440	32
		52	3.90			416	
E6 (P2E1)	8	168	4.03	16	0.016	1344	128
		52	3.70			416	
F0	4	184	4.03	0	——	736	0
		52	3.80			208	
F1	24	180	3.80	4	0.014	4320	96
		52	3.50			1248	
F2	4	172	3.80	12	0.026	688	48
		52	3.50			208	
F3	8	176	3.80	8	0.022	1408	64
		52	3.50			416	
F4	16	168	4.03	16	0.028	2688	256
		52	3.50			832	
F5	32	172	4.03	12	0.026	5504	384
		52	3.50			1664	
Total	241		3.85			54700	2176

FIGURE 3-1

PALO VERDE UNIT 1 CYCLE 4

ASSEMBLIES TO BE DISCHARGED AT EOC-3
FUEL AND BURNABLE POISON ROD PLACEMENT

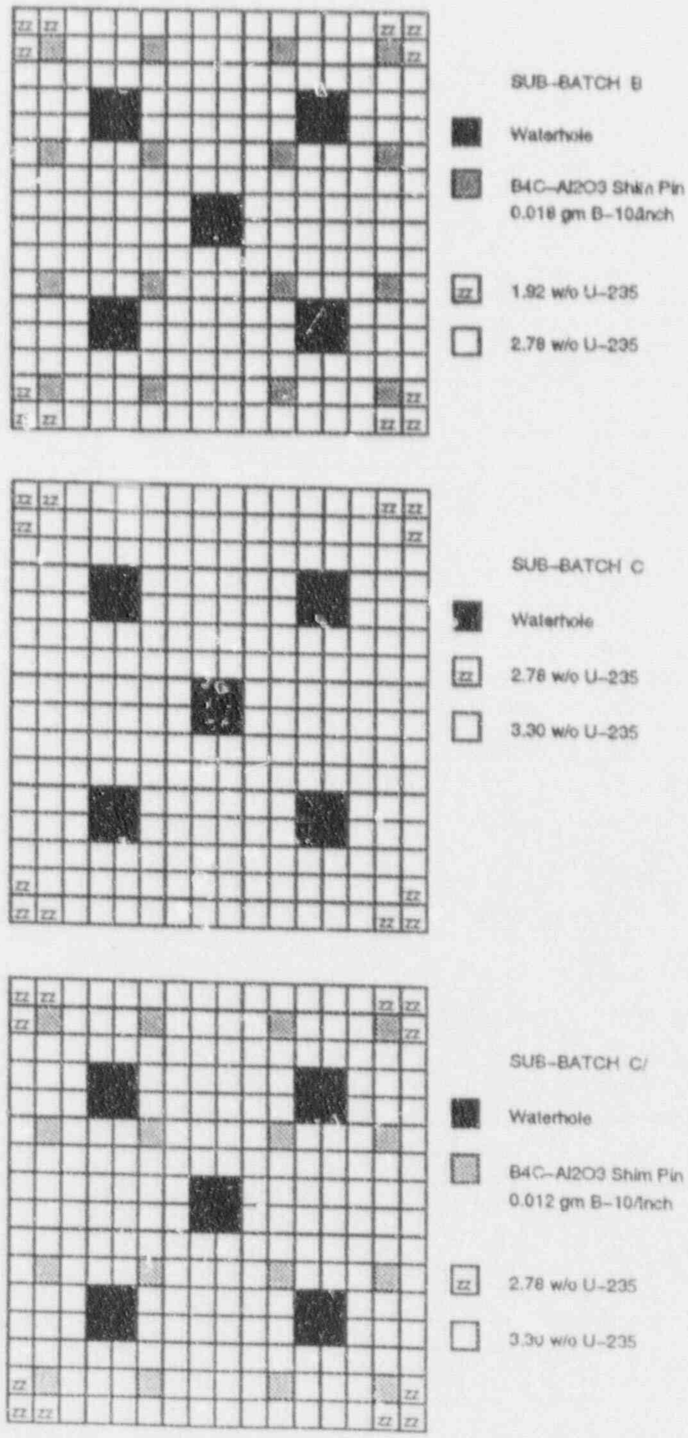


FIGURE 3-1 (Continued)

PALO VERDE UNIT 1 CYCLE 4

ASSEMBLIES TO BE DISCHARGED AT EOC-3
FUEL AND BURNABLE POISON ROD PLACEMENT

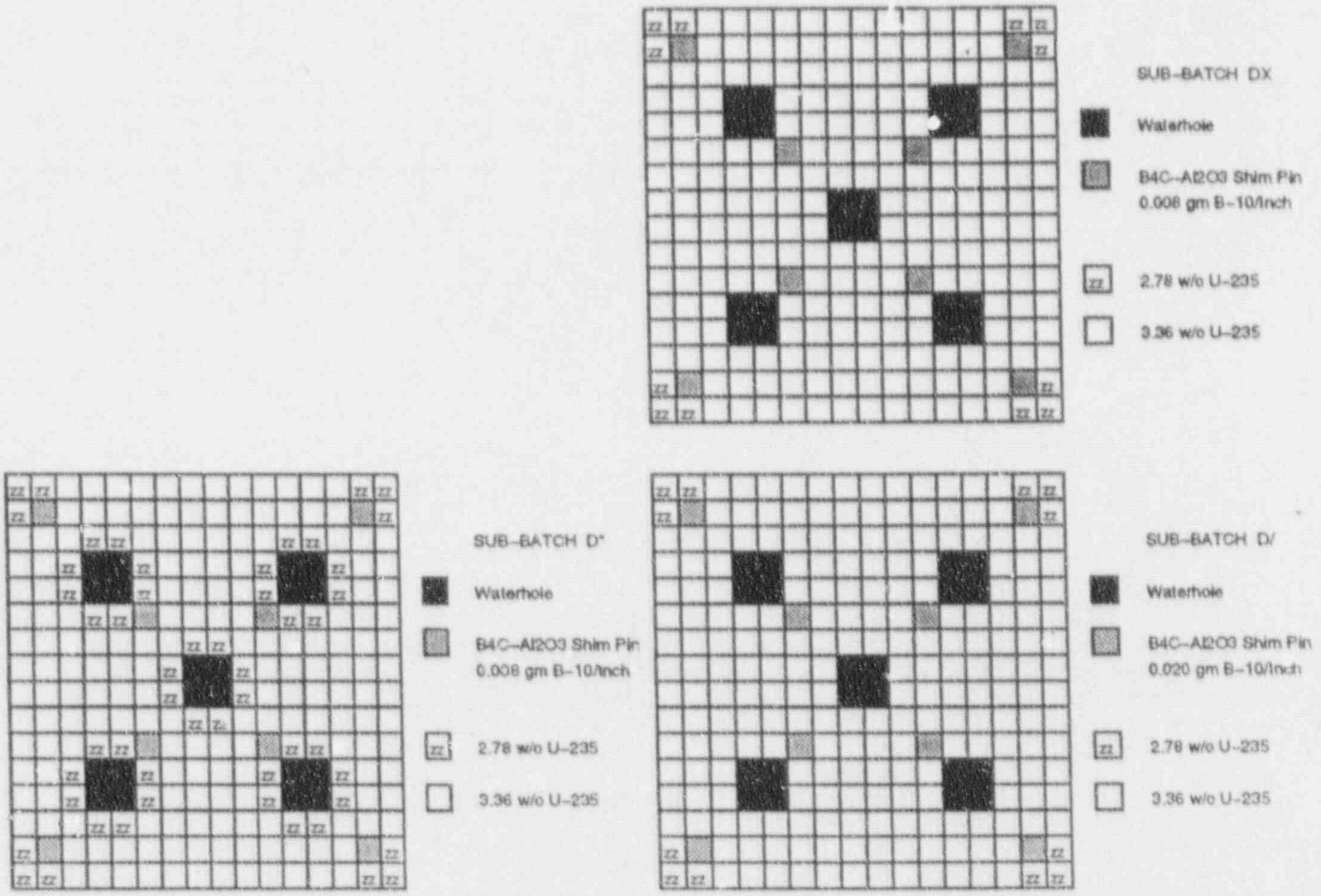
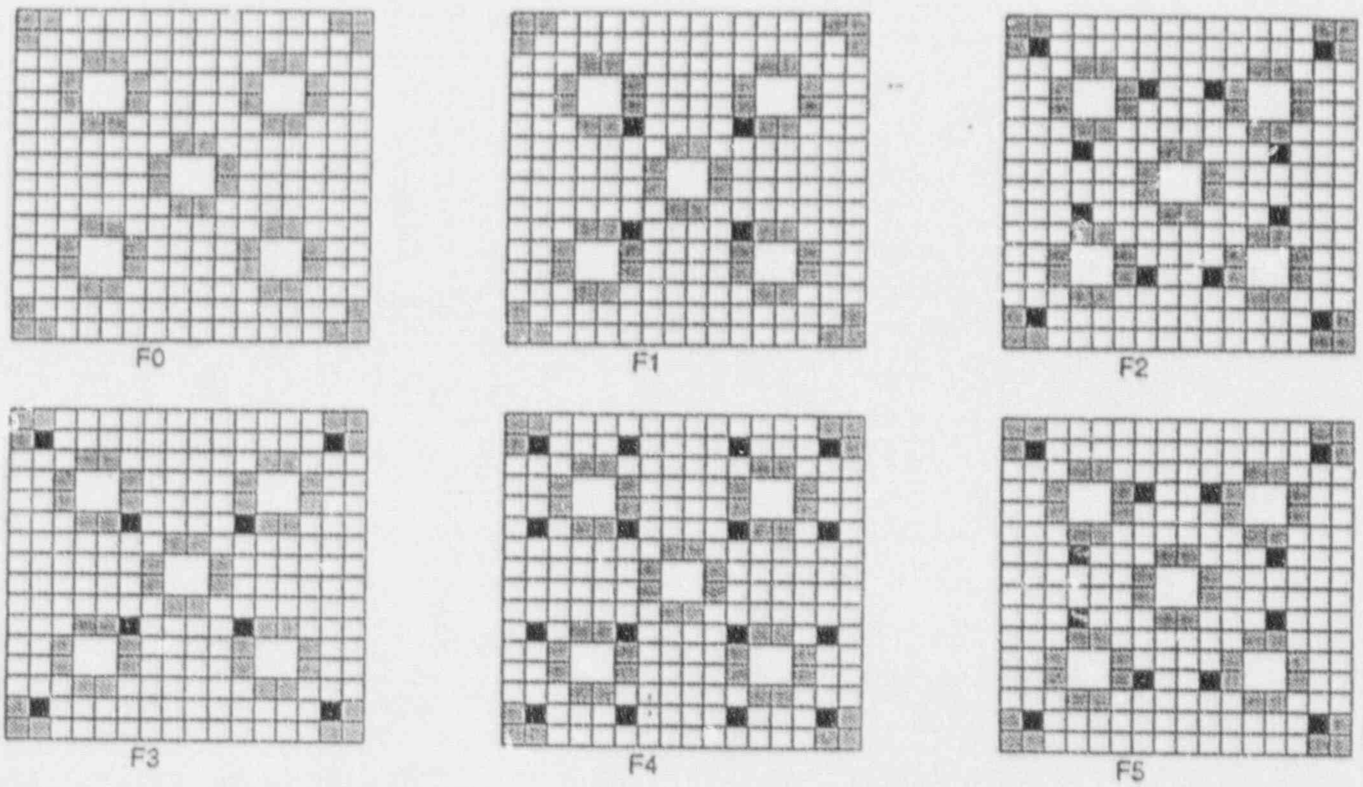


FIGURE 3-2

PALO VERDE UNIT 1 CYCLE 4

FRESH FEED FUEL ASSEMBLIES
FUEL AND BURNABLE POISON ROD PLACEMENT



FUEL TYPE	No. OF ASSEMBLIES	ENRICHMENT W/O □	ENRICHMENT W/O ▨	No. OF SHIMS ■	GM B-10/IN
F0	4	4.03	3.80	0	-------
F1	24	3.80	3.50	4	0.014
F2	4	3.80	3.50	12	0.026
F3	8	3.80	3.50	8	0.022
F4	16	4.03	3.50	16	0.028
F5	32	4.03	3.50	12	0.026

					1 E0	2 E0	3 E4	4 F0
			5 D	6 E1	7 F1	8 E0	9 F1	10 C
		11 E2	12 E6	13 F1	14 D	15 F5	16 D	17 F4
	18 D	19 E6	20 F3	21 E4	22 F4	23 D	24 F5	25 E1
	26 E1	27 F1	28 E4	29 F2	30 E1	31 F5	32 E3	33 E5
34 E0	35 F1	36 D	37 F4	38 E1	39 E3	40 E2	41 F5	42 B
43 E0	44 E0	45 F5	46 D	47 F5	48 E2	49 F4	50 E4	51 F3
52 E4	53 F1	54 D	55 F5	56 E3	57 F5	58 E4	59 D	60 E5
61 F0	62 C	63 F4	64 E1	65 E5	66 B	67 F3	68 E5	69 B

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 CORE MAP	FIGURE 3-3

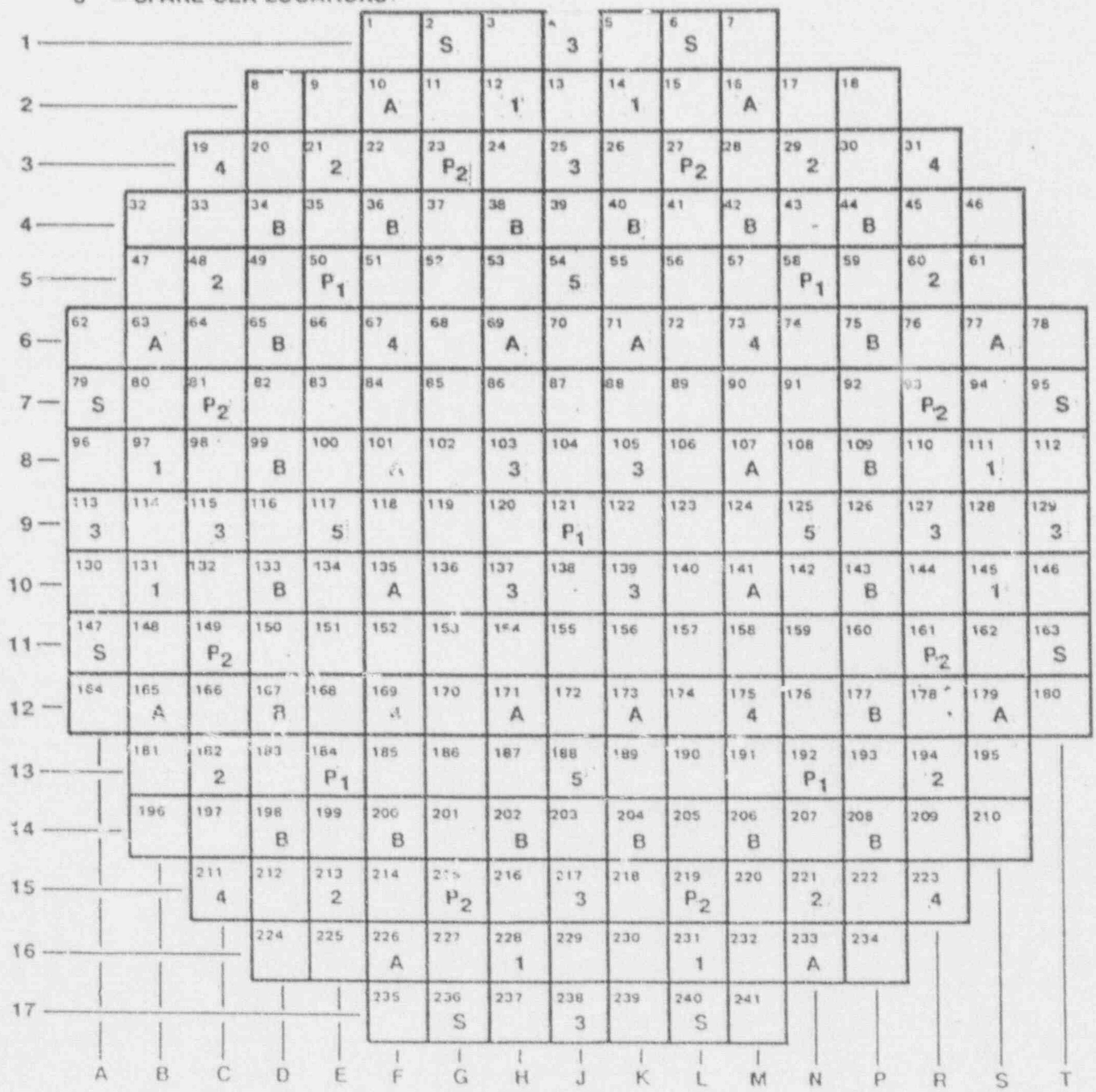
AA BB
CC

AA = Quarter Core Location
BB = Batch Type
CC = BOC-4 Burnup (MWD/T),
EOC-3=517 EFPD

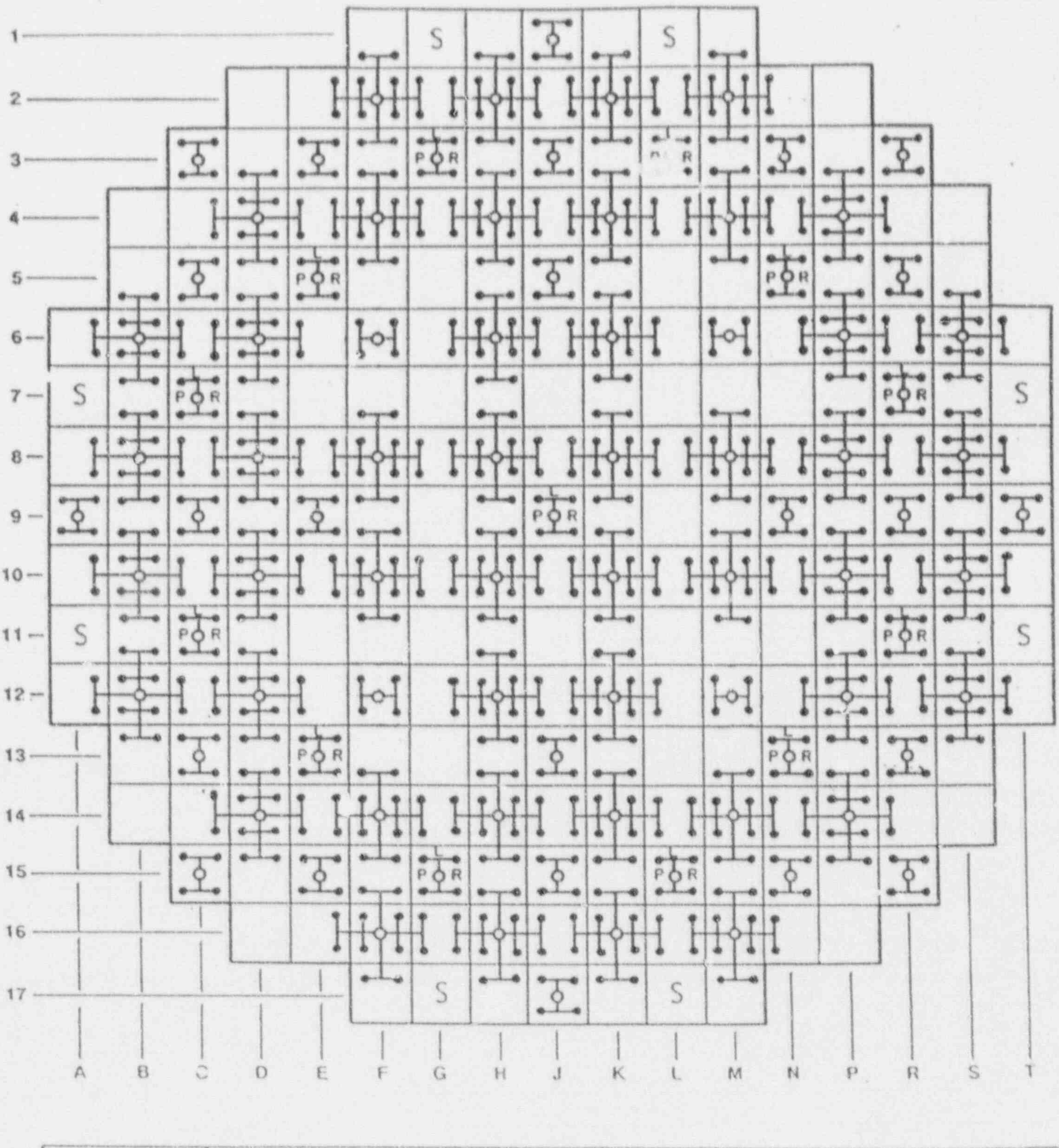
					1 E0	2 E0	3 E4	4 F0
					17,268	14,683	23,171	0
			5 D	6 E1	7 F1	8 E0	9 F1	10 C
			22,637	25,149	0	17,209	0	24,188
		11 E2	12 E6	13 F1	14 D	15 F5	16 D	17 F4
		24,311	19,766	0	31,578	0	33,858	0
	18 D	19 E6	20 F3	21 E4	22 F4	23 D	24 F5	25 E1
	22,640	19,762	0	23,645	0	35,054	0	25,851
	26 E1	27 F1	28 E4	29 F2	30 E1	31 F5	32 E3	33 E5
	25,160	0	23,639	0	25,456	0	26,111	21,020
34 E0	35 F1	36 D	37 F4	38 E1	39 E3	40 E2	41 F5	42 B
17,257	0	31,561	0	25,458	25,456	26,073	0	19,717
43 E0	44 E0	45 F5	46 D	47 F5	48 E2	49 F4	50 E4	51 F3
14,677	17,204	0	35,002	0	26,091	0	25,652	0
52 E4	53 F1	54 D	55 F5	56 E3	57 F5	58 E4	59 D	60 E5
23,167	0	33,848	0	26,129	0	25,619	32,561	17,825
61 F0	62 C	63 F4	64 E1	65 E5	66 B	67 F3	68 E5	69 B
0	24,188	0	25,851	21,020	19,717	0	17,825	19,970

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY AVERAGE BURNUP BOC	FIGURE 3-4

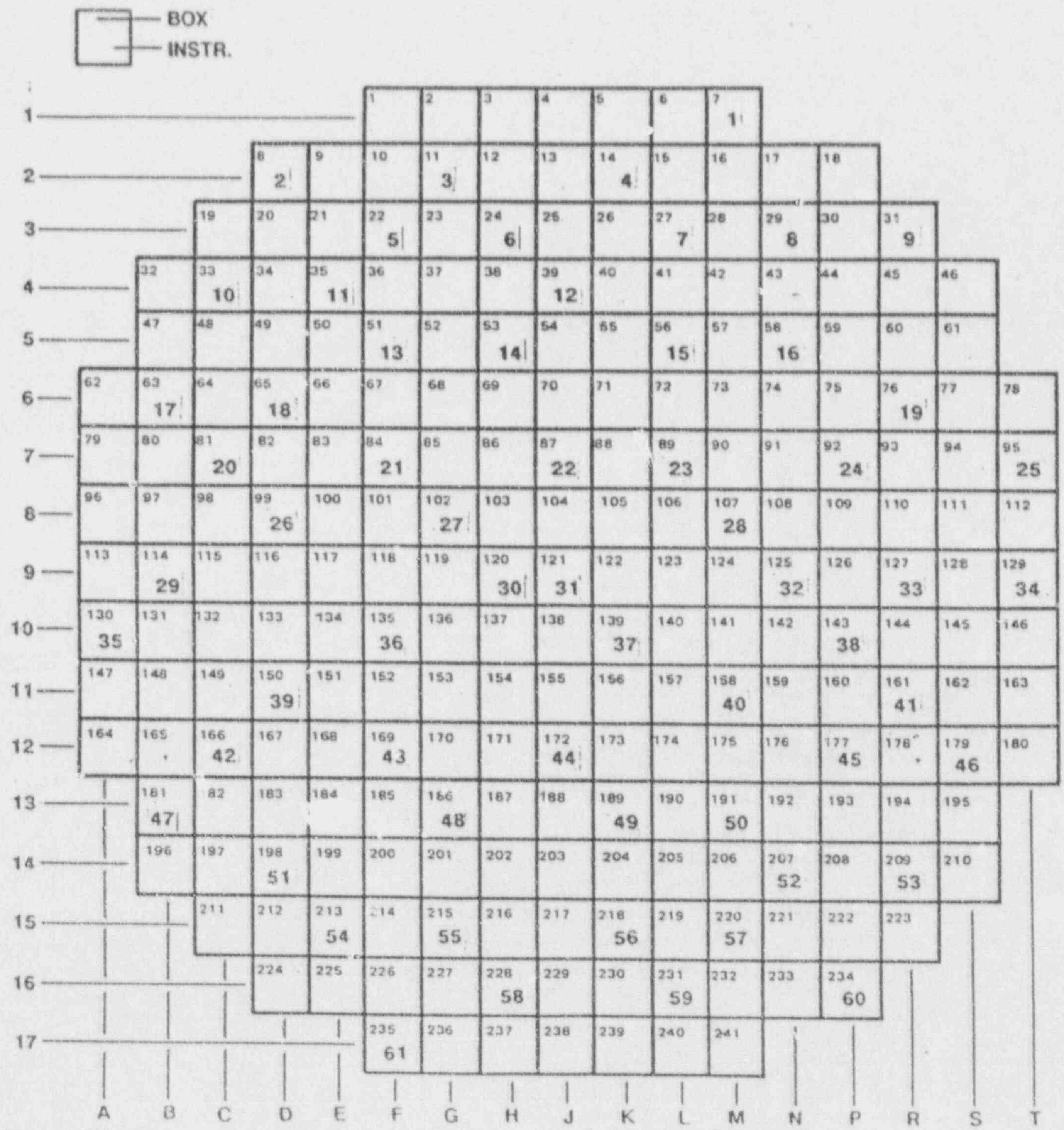
5 – LEAD REGULATING BANK
4 – SECOND REGULATING BANK
3 – THIRD REGULATING BANK
2 – FOURTH REGULATING BANK
1 – LAST REGULATING BANK
B – SHUTDOWN BANK B
A – SHUTDOWN BANK A
P_2 – PLR GROUP 2
P_1 – PLR GROUP 1
S – SPARE CEA LOCATIONS



ARIZONA Palo Verde Nuclear Generating Station	CEA BANK IDENTIFICATION	Figure 3-5A



ARIZONA Palo Verde Nuclear Generating Station	CEA PATTERN	Figure 3-58



ARIZONA Palo Verde Nuclear Generating Station	INSTRUMENT LOCATIONS	Figure 3-6

4.0 FUEL SYSTEM DESIGN

4.1 MECHANICAL DESIGN

4.1.1 Fuel Design

The mechanical design of the Batch F reload fuel assemblies is identical to the design of the Reference Cycle Batch E reload fuel assemblies except for a modification to the lower end fitting and center guide tube design. No changes in the mechanical design bases have occurred since the original fuel design.

The following design features were incorporated into Batch F.

The lower end fitting design was changed from a two piece assembly to a single piece casting with a recess for the center guide tube to fit within the flow plate.

The length of the center guide tube was increased from 163.715 inches to 163.965 inches in order to fit within the new lower end fitting.

The new design provides improved strength, stiffness, and quality in the lower end fitting.

4.2 GUIDE TUBE WEAR

Twenty of the fuel assemblies that had CEAs located in them during Cycle 1 at Palo Verde Unit 1 were inspected for guide tube wear. That inspection was part of the required licensing procedures required by the NRC for all plants after the first cycle of operation (References 4-1, 4-7, and 4-8). A similar program was also performed on Unit 2 during the first refueling outage (Reference 4-2 and 4-6). The number of assemblies inspected for guide tube wear was determined based on the results of the Unit 1 inspection. The inspections revealed that guide tube wear was minor

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4.3 THERMAL DESIGN

The thermal performance of composite fuel pins that envelope the pins of fuel batches B, C, D, E and F present in Cycle 4 has been evaluated using the FATES-3A version of the C-E fuel evaluation model (References 4-3 and 4-4). The analysis was performed using a power history that enveloped the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at the end of Cycle 4. The rod internal pressure remains below the reactor coolant pressure throughout Cycle 4. The power to centerline melt limit has been determined to be in excess of 21 kW/ft.

4.4 CHEMICAL DESIGN

The metallurgical requirements of the fuel cladding for the Batch F fuel assemblies are the same for the Batch E assemblies. The metallurgical requirements of the fuel assembly structural members for the Batch F are the same as the Batch E fuel batches included in Cycle 3. Thus the chemical metallurgical performance of the Batch F fuel will be similar to (or better than) the Batch E fuel used in Cycle 3.

4.5 SHOULDER GAP ADEQUACY

The present shoulder gap is projected to be adequate for Cycle 4 operation. This conclusion is based on the fuel rod growth models of Reference 4-9 in conjunction with the measurements conducted post Unit 1 Cycle 2, Reference 4-1.

5.0 NUCLEAR DESIGN

5.1 PHYSICS CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 4 core makes use of a low-leakage fuel management scheme, in which previously burned assemblies are placed on the core periphery. Most of the fresh Batch F assemblies are located throughout the interior of the core where they are mixed with the previously burned fuel in a pattern that minimizes power peaking. With this loading and a Cycle 3 endpoint of 491 EFPD, the Cycle 4 reactivity lifetime for full power operation is expected to be 400 EFPD. Explicit evaluations have been performed to assure applicability of all analyses to a Cycle 3 termination burnup of between 465 and 517 EFPD and for a Cycle 4 length up to 426 EFPD.

Characteristic physics parameters for Cycle 4 are compared to those of the Reference Cycle in Table 5-1. The values in this table are intended to represent nominal core parameters. Those values used in the safety analysis (see Sections 7 and 8) contain appropriate uncertainties, or incorporate values to bound future operating cycles, and in all cases are conservative with respect to the values reported in Table 5-1.

Table 5-2 presents a summary of CEA reactivity worths and allowances for the end of Cycle 4 full power steam line break transient with a comparison to the Reference Cycle data. The full power steam line break was chosen to illustrate differences in CEA reactivity worths for the two cycles.

The CEA core locations and group identifications remain the same as in the Reference Cycle. The power dependent insertion limit (PDIL) for regulating groups and part length CEA groups is shown in Figures 5-1 and 5-2, respectively. Table 5-3 shows the reactivity worths of various regulating CEA groups calculated at full power conditions for Cycle 4 and the Reference Cycle.

5.1.2 Power Distributions

Figures 5-3 through 5-5 illustrate the calculated All Rods Out (ARO) relative assembly power densities during Cycle 4. The one-pin planar radial power peaks (F_{xy}) presented in these figures represent the maximum over the mid eighty percent of the core's axial height. Time points at the beginning, middle, and end of cycle were chosen to display the variation in assembly and maximum planar radial peaking as a function of burnup.

Relative assembly power densities for rodded configurations are given for BOC and EOC in Figures 5-6 through 5-11. The rodded configuration shown are those allowed by the PDIL at full power: part length CEAs (PLCEAs), Bank 5, and Bank 5 plus the PLCEAs.

The radial power distributions described in this section are calculated data which do not include any uncertainties or allowances. The calculations performed to determine these radial power peaks explicitly account for augmented power peaking which is characteristic of fuel rods adjacent to the water holes.

Nominal axial peaking factors are expected to range from 1.16 at BOC4 to 1.08 at EOC4.

5.2 PHYSICS ANALYSIS METHODS

5.2.1 Analytical Input to In-Core Measurements

In-core detector measurement constants to be used in evaluating the reload cycle power distributions will be calculated in accordance with Reference 5-1. The ROCS and MC codes employing DIT calculated cross sections will be used. ROCS, MC, and DIT have been approved for this application in Reference 5-2.

5.2.2 Uncertainties in Measured Power Distributions

The planar radial power distribution measurement uncertainty of 5.3%, based on Reference 5-1, will be applied to the Cycle 4 COLSS and CPC on-line calculations which use planar radial power peaks. The axial and three dimensional power distribution measurement uncertainties are determined in conjunction with other monitoring and protection system measurement uncertainties, as was done for Cycle 3.

5.2.3 Nuclear Design Methodology

The Cycle 4 nuclear design was performed using the DIT, ROCS, and MC computer codes described in Reference 5-2 with the minor improvements described below. In addition, the Appendix to this report contains the 50.59 determination that use of these improved codes does not require explicit NRC review.

5.2.3.1 Nuclear Design Code Improvements

Over the past several years, ABB Combustion Engineering Nuclear Power (CENP) has improved the codes and methods used to analyze NSSS and reload fuel designs. Most of the code improvements fall in the categories of improved calculational efficiency, improved user friendliness, and improved exchange of data between different code modules. Only four of the improvements affect the calculational accuracy of the results. These four improvements (addition of the nodal expansion method, anisotropic scattering, higher order interface currents, and assembly discontinuity factors) have been demonstrated to result in improved accuracy. In addition to the incorporation of these improvements, the associated biases and uncertainties were revised as part of the overall verification process to insure that 95/95 confidence limits are maintained in all licensing related calculations.

5.2.3.1.1 Nodal Expansion Method

The use of Nodal Expansion Solution Method (NEM) in the ROCS code was discussed in the original CENP ROCS/DIT Topical Report (Reference 5-2) even though it had not yet been fully integrated into the ROCS computer code. Recognizing this fact, the NRC stipulated only that CENP ensure that equivalent biases and uncertainties be obtained when NEM is incorporated into the ROCS code.

Prior to implementation of this improvement to the nuclear design code, CENP performed numerous benchmark calculations using data from past reload cycles. Updated calculational biases and uncertainties were still defined by the 95/95 confidence limits. Equivalence was, thus, maintained and the limitation of the NRC's approval of the Topical Report has not been violated.

5.2.3.1.2 Anisotropic Scattering and Higher Order Interface Currents

The use of Anisotropic Scattering and Higher Order Interface Currents in the DIT code were discussed in CENP's Gadolinia-Urania Topical Report (Reference 5-3). In the approval of the report the NRC stated:

"We have reviewed the Combustion Engineering Licensing Topical Report CENPD-275-P, Revision 1-P. Based on our review, we conclude that the gadolinia fuel properties are acceptable for licensing applications up to 8 weight percent gadolinia concentration. We also conclude that the neutronics methods described in the report (DIT, ROCS/MC and PDQ), as modified, are acceptable for calculating the neutronic characteristics of PWR cores containing up to 8 weight percent gadolinia bearing fuel rods."

Since the analysis presented in Reference 5-3 included assemblies which contained B_4C poison rods or no poison material at all, the case of zero percent gadolinia is included in the range of applicability.

5.2.3.1.3 Assembly Discontinuity Factors

Use of Assembly Discontinuity Factors (ADFs) in ROCS differs from the improvements discussed above in that the function of the ADFs are to improve the internal agreement between two existing modules of the approved code system (ROCS and DIT). Furthermore, unlike the other methods of improvement, where improved accuracy must be demonstrated by statistical analysis of measured to calculated errors, the improvement of internal agreement resulting from the addition of ADFs can be verified at any time simply by comparing the ROCS and DIT computer output for the case of interest. It is the opinion of APS that the addition of ADFs has not changed the overall code system representation of reality. Their use, as a significant and widely utilized industry breakthrough in PWR calculational ability, is documented in Reference 5-4.

5.2.3.2 Revised Biases and Uncertainties

Implementation of the improved methods has necessitated an update of the biases and uncertainties used to assure that 95/95 confidence limits are maintained in all results used for licensing related analyses. The revised biases and uncertainties were established by comparing results obtained from analytical calculation with measured data. The re-evaluation of biases and uncertainties used the same statistical methodology (with the exception of the N-1 rod worth as discussed below) as described in the ROCS/DIT Topical Report (Reference 5-2). Consequently, CENP has concluded that the new biases and uncertainties fall within the original basis for acceptance of the ROCS/DIT Topical Report by the NRC in so much as the results are judged to be equivalent when compared to other biases

In the ROCS/DIT Topical Report (Reference 5-2), the bias and uncertainty associated with net (N-1) rod worth is explicitly calculated by evaluating the net rod worth measurements performed during intial core startups. These evaluations showed a 3.6% underprediction of the N-1 rod worth, with a 1.47% standard deviation about the mean value. This standard deviation is quite small and was deemed inappropriate for use in reload analysis for two reasons. First, the N-1 statistics were based on a small number of N-1 rod worth measurements performed. Second, the N-1 measurements were taken during the beginning of cycle for the initial cores, and hence may not be fully representative of later cycles. In view of these limitations of the N-1 statistics, the Topical Report embraced a conservative approach which applied the bias and uncertainty associated with individual bank worth to the N-1 rod worth.

It is recognized, however, that using the uncertainty for an individual bank for the N-1 rod worth is overly conservative. This is true because the maximum individual rod uncertainty is often dominated by rod banks with low worths. For low worth rod banks, the percentage uncertainty is often high despite the fact that the absolute value of the uncertainty is small and well within the experimental precision.

CENP has, hence, re-evaluated the bias and uncertainty for the N-1 configuration. In particular, the N-1 bias and uncertainty used are the bias and uncertainty associated with the sum of the bank worths (i.e., "total" worth). The use of the total rod worth uncertainty is considered more appropriate than the individual bank worth since the total rod worth configuration is more representative of the higher control rod density of the N-1 configuration.

This alternative is still conservative because actual N-1 measurements indicate that the uncertainty of the N-1 rod worth is really lower than the uncertainty of the total worth. CENP has performed calculations which demonstrate that the N-1 configuration is strongly influenced by the reactivity of the unrodded region of the core. Thus, the N-1 configuration is less sensitive to the precision of the calculated effective control rod cross section as compared to either the total or individual bank configurations.

This approach is consistent with the assumption in the Topical Report in which the total worth and N-1 rod configuration are assumed to belong to the same population. Thus, it is considered that the approach for the N-1 case yields equivalent calculational biases and uncertainties as compared to similar quantities calculated using the nuclear design codes and methods described in the Topical Report.

TABLE 5-1

PVNGS UNIT 1 CYCLE 4

NOMINAL PHYSICS CHARACTERISTICS

DESCRIPTION	UNITS	REFERENCE CYCLE	CYCLE 4
Dissolved Boron	PPM		
Dissolved Boron Concentration for Criticality, CEAs Withdrawn Hot Full Power, Equilibrium Xenon BOC		1223	1120
Boron Worth	PPM/%Δρ		
Hot Full Power, BOC		127	128
Hot Full Power, EOC		98	100
Moderator Temperature Coefficients	10^{-4} Δρ/°F		
Hot Full Power, Equilibrium Xenon			
BOC		-0.6	-1.05
EOC		-3.3	-3.30
Hot Zero Power, BOC		+0.3	+0.02
Doppler Coefficients	10^{-5} Δρ/°F		
Hot Zero Power, BOC		-2.1	-1.81
Hot Full Power, BOC		-1.7	-1.52
Hot Full Power, EOC		-1.9	-1.64
Total Delayed Neutron Fraction, βeff	-----		
BOC		0.0069 (1)	0.0061
EOC		0.0046 (1)	0.0051
Prompt Neutron Generation Time, ℓ*	10^{-6} sec		
BOC		20.7	20.1
EOC		27.3	25.8

(1) The differences between these reference cycle values and those presented for Cycle 4 are due to the inclusion of an uncertainty. Removing this uncertainty from the reference cycle data yields Total Delayed Neutron Fractions similar to those of Cycle 4. Thus, the reference cycle values, calculated without the uncertainty, are as follows:

i) BOC = 0.0063
ii) EOC = 0.0051

TABLE 5-2

PVNGS UNIT 1 CYCLE 4
LIMITING VALUES OF REACTIVITY WORTHS
AND ALLOWANCES FOR HOT FULL POWER STEAM LINE BREAK, %$\Delta\rho$
END-OF-CYCLE (EOC)

	DESCRIPTION	REFERENCE CYCLE	CYCLE 4
1.	Worth of all CEA's Inserted	-18.0	-16.2*
2.	Stuck CEA Allowance	+5.5	+3.9
3.	Worth of all CEAs Less highest Worth CEA Stuck Out	-12.5	-12.3*
4.	Full Power Dependent Insertion Limit CEA Bite	+0.2	+0.3
5.	Calculated Scram Worth	-12.3	-12.0
6.	Physics Uncertainty	+1.2	+0.8*
7.	Other Allowances	+0.1	+0.1
8.	Net Available Scram Worth	-11.0	-11.1
9.	Scram Worth Used in Safety Analysis	-10.2	-10.2

* Deviation in the Cycle 4 values of Items 1, 3 and 6 from those given for the reference cycle are due to the effects of fuel management differences and improvements in nuclear design methodology.

TABLE 5-3

PVNGS UNIT 1 CYCLE 4

REACTIVITY WORTH OF CEA REGULATING GROUPS
AT HOT FULL POWER, %Δρ

REGULATING CEAs	BEGINNING OF CYCLE		END OF CYCLE	
	REFERENCE CYCLE	CYCLE 4	REFERENCE CYCLE	CYCLE 4
Group 5	-0.31	-0.26	-0.33	-0.28
Group 4	-0.37	-0.29	-0.39	-0.34
Group 3	-0.91	-0.78	-0.92	-0.87

Notes:

Values shown assume sequential group insertion.

FIGURE 5-1

CEA INSERTION LIMITS VS. THERMAL POWER (COLSS IN SERVICE)

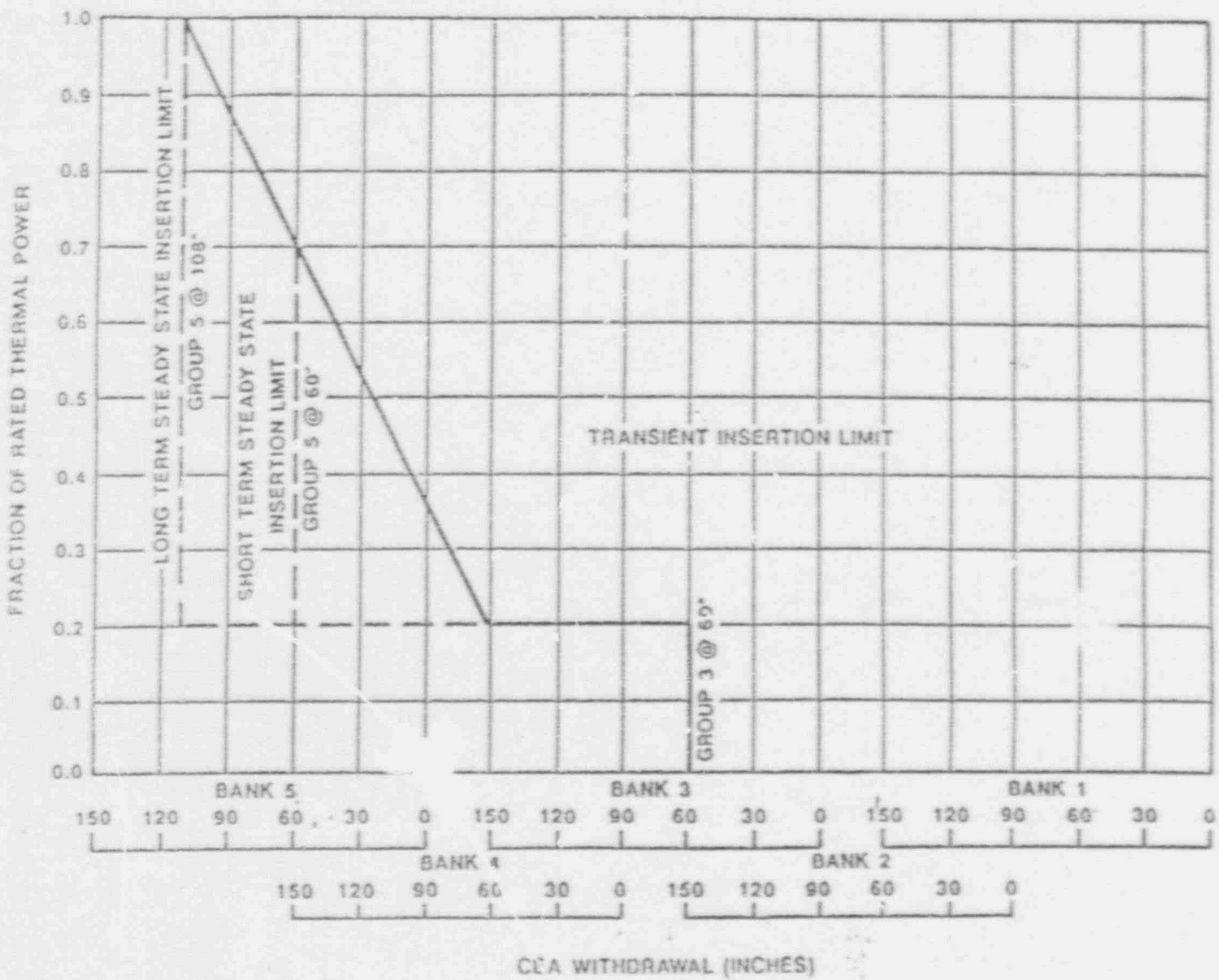
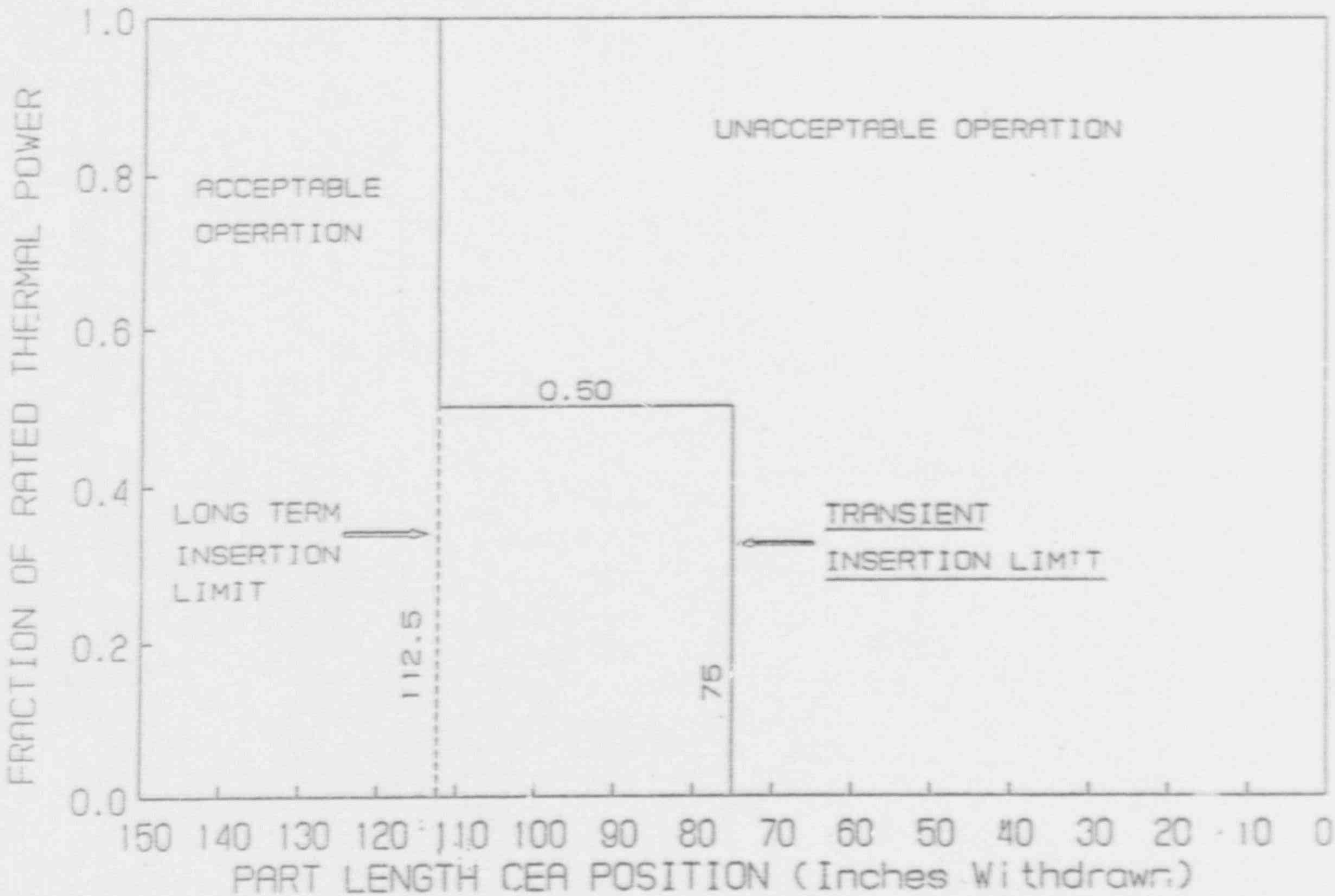


FIGURE 5-2

PART LENGTH CEA, POSITION VS. THERMAL POWER



AA BB CC	AA = Quarter Core Location BB = Batch Type CC = Relative Power Density

					1 E0 0.47	2 E0 0.64	3 E4 0.65	4 F0 0.84
			5 D 0.35	6 E1 0.64	7 F1 1.08	8 E0 1.11	9 F1 1.24	10 C 0.88
		11 E2 0.40	12 E6 0.80	13 F1 1.19	14 D 0.94	15 F5 1.28	16 D 0.94	17 F4 1.21
	18 D 0.35	19 E6 0.80	20 F3 1.23	21 E4 1.13	22 F4 1.23	23 D 0.91	24 F5 1.27	25 E1 1.09
	26 E1 0.64	27 F1 1.19	28 E4 1.14	29 F2 1.30	30 E1 1.10	31 F5 1.28	32 E3 1.09	33 E5 1.10
34 E0 0.47	35 F1 1.08	36 D 0.94	37 F4 1.24	38 E1 1.10	39 E3 1.04	40 E2 1.06	41 F5 1.30	42 B 0.96
43 E0 0.65	44 E0 1.11	45 F5 1.28	46 D 0.91	47 F5 1.28	48 E2 1.06	49 F4 1.22	50 E4 1.06	51 F3 1.32
52 E4 0.65	53 F1 1.25 X	54 D 0.94	55 F5 1.27	56 E3 1.09	57 F5 1.30	58 E4 1.06	59 D 0.90	60 E5 1.11
61 F0 0.84	62 C 0.90	63 F4 1.21	64 E1 1.09	65 E5 1.11	66 B 0.96	67 F3 1.32	68 E5 1.11	69 B 0.88

NOTE: X = MAXIMUM F_{xy} = 1.52

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES AT BOC, ARO, HFP, Eq XE	FIGURE 5-3

AA BB CC	AA = Quarter Core Location BB = Batch Type CC = Relative Power Density

					1 E0 0.47	2 E0 0.63	3 E4 0.63	4 F0 0.80
			5 D 0.37	6 E1 0.64	7 F1 1.05	8 E0 1.04	9 F1 1.19	10 C 0.86
		11 E2 0.43	12 E6 0.81	13 F1 1.17	14 D 0.93	15 F5 1.29	16 D 0.94	17 F4 1.26
	18 D 0.37	19 E6 0.81	20 F3 1.23	21 E4 1.11	22 F4 1.30	23 D 0.94	24 F5 1.31	25 E1 1.09
	26 E1 0.64	27 F1 1.17	28 E4 1.11	29 F2 1.34 X	30 E1 1.11	31 F5 1.34	32 E3 1.09	33 E5 1.09
34 E0 0.47	35 F1 1.05	36 D 0.93	37 F4 1.30	38 E1 1.11	39 E3 1.05	40 E2 1.07	41 F5 1.34	42 B 0.96
43 E0 0.63	44 E0 1.04	45 F5 1.29	46 D 0.94	47 F5 1.34	48 E2 1.07	49 F4 1.28	50 E4 1.04	51 F3 1.30
52 E4 0.63	53 F1 1.19	54 D 0.94	55 F5 1.31	56 E3 1.09	57 F5 1.34	58 E4 1.04	59 D 0.87	60 E5 1.04
61 F0 0.80	62 C 0.87	63 F4 1.26	64 E1 1.09	65 E5 1.09	66 B 0.96	67 F3 1.30	68 E5 1.04	69 B 0.84

NOTE: X = MAXIMUM F_{xy} = 1.45

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES AT MOC, ARO, HFP, Eq XE	FIGURE 5-4

AA BB CC	AA = Quarter Core Location BB = Batch Type CC = Relative Power Density

					1 E0 0.48	2 E0 0.64	3 E4 0.63	4 F0 0.79
			5 D 0.40	6 E1 0.65	7 F1 1.03	8 E0 1.00	9 F1 1.15	10 C 0.86
		11 E2 0.46	12 E5 0.81	13 F1 1.14	14 D 0.93	15 F5 1.32	16 D 0.95	17 F4 1.34
	18 D 0.40	19 E6 0.81	20 F3 1.20	21 E4 1.08	22 F4 1.37 X	23 D 0.97	24 F5 1.35	25 E1 1.09
	26 E1 0.65	27 F1 1.13	28 E4 1.08	29 F2 1.35	30 E1 1.10	31 F5 1.36	32 E3 1.07	33 E5 1.05
34 E0 0.48	35 F1 1.03	36 D 0.93	37 F4 1.37	38 E1 1.10	39 E3 1.03	40 E2 1.06	41 F5 1.34	42 B 0.94
43 E0 0.64	44 E0 1.00	45 F5 1.32	46 D 0.97	47 F5 1.36	48 E2 1.06	49 F4 1.34	50 E4 1.02	51 F3 1.24
52 E4 0.63	53 F1 1.15	54 D 0.95	55 F5 1.35	56 E3 1.07	57 F5 1.34	58 E4 1.02	59 D 0.84	60 E5 0.97
61 F0 0.79	62 C 0.87	63 F4 1.34	64 E1 1.09	65 E5 1.06	66 B 0.95	67 F3 1.24	68 E5 0.97	69 B 0.80

NOTE: X = MAXIMUM F_{xy} = 1.48

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES AT EOC, ARO, HFP, Eq XE	FIGURE 5-5

AA BB CC PL

AA = Quarter Core Location
BB = Batch Type
CC = Relative Power Density
PL = Part Length CEA Location

					1 E0 0.46	2 E0 0.63	3 E4 0.64	4 F0 0.86
			5 D 0.34	6 E1 0.63	7 F1 1.07	8 E0 1.08	9 F1 1.26	10 C 0.88
		11 E2 0.39	12 E6 0.79	13 F1 1.20	14 D 0.90	15 F5 1.14 PL	16 D 0.91	17 F4 1.23
	18 D 0.34	19 E6 0.79	20 F3 1.24	21 E4 1.10	22 F4 1.21	23 D 0.88	24 F5 1.29	25 E1 1.10
	26 E1 0.63	27 F1 1.20	28 E4 1.10	29 F2 1.15 PL	30 E1 1.07	31 F5 1.32	32 E3 1.12	33 E5 1.15
34 E0 0.46	35 F1 1.08	36 D 0.91	37 F4 1.21	38 E1 1.07	39 E3 1.04	40 E2 1.09	41 F5 1.39	42 B 0.99
43 E0 0.63	44 E0 1.09	45 F5 1.14 PL	46 D 0.88	47 F5 1.32	48 E2 1.09	49 F4 1.29	50 E4 1.12	51 F3 1.43 X
52 E4 0.65	53 F1 1.27	54 D 0.92	55 F5 1.30	56 E3 1.12	57 F5 1.40	58 E4 1.11	59 D 0.93	60 E5 1.15
61 F0 0.86	62 C 0.90	63 F4 1.23	64 E1 1.10	65 E5 1.16	66 B 0.99	67 F3 1.43	68 E5 1.15	69 B 0.81 PL

NOTE: X = MAXIMUM F_{xy} = 1.56

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES BOC, PLCEA'S INSERTED, HFP, ARO EQ. XENON	FIGURE 5-6

AA BB CC BK5	AA = Quarter Core Location BB = Batch Type CC = Relative Power Density BK5 = CEA Bank 5 Location

					1 E0 0.50	2 E0 0.68	3 F4 0.68	4 F0 0.90
			5 D 0.37	6 E1 0.69	7 F1 1.18	8 E0 1.19	9 F1 1.33	10 C 0.91
		11 E2 0.42	12 E6 0.87	13 F1 1.31	14 D 1.00	15 F5 1.35	16 D 0.94	17 F4 1.21
	18 D 0.37	19 E6 0.87	20 F3 1.37 X	21 E4 1.22	22 F4 1.30	23 D 0.89	24 F5 1.20	25 E1 0.95
	26 E1 0.69	27 F1 1.32	28 E4 1.22	29 F2 1.39	30 E1 1.11	31 F5 1.25	32 E3 0.93	33 E5 0.69 BK5
34 E0 0.50	35 F1 1.18	36 D 1.00	37 F4 1.30	38 E1 1.11	39 E3 1.01	40 E2 1.00	41 F5 1.18	42 B 0.79
43 E0 0.68	44 E0 1.19	45 F5 1.36	46 D 0.90	47 F5 1.25	48 E2 1.00	49 F4 1.15	50 E4 0.97	51 F3 1.23
52 E4 0.68	53 F1 1.34	54 D 0.94	55 F5 1.20	56 E3 0.93	57 F5 1.18	58 E4 0.97	59 D 0.82	60 E5 1.03
61 F0 0.90	62 C 0.93	63 F4 1.21	64 E1 0.95	65 E5 0.69 BK5	66 B 0.79	67 F3 1.23	68 E5 1.02	69 B 0.80

NOTE: X = MAXIMUM F_{xy} = 1.59

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES BOC, BANK 5 INSERTED, HFP, ARO EQ. XENON	FIGURE 5-7

AA BB CC BK5 PL

AA = Quarter Core Location
BB = Batch Type
CC = Relative Power Density
PL = Part Length CEA Location
BK5 = Bank 5 CEA Location

					1 E0 0.49	2 E0 0.68	3 E4 0.68	4 F0 0.91
			5 D 0.37	6 E1 0.68	7 F1 1.16	8 E0 1.15	9 F1 1.32	10 C 0.92
		11 E2 0.42	12 E6 0.87	13 F1 1.30	14 D 0.96	15 F5 1.18 PL	16 D 0.92	17 F4 1.21
	18 D 0.37	19 E6 0.87	20 F3 1.35 X	21 E4 1.18	22 F4 1.27	23 D 0.88	24 F5 1.20	25 E1 0.97
	26 E1 0.68	27 F1 1.30	28 E4 1.18	29 F2 1.21 PL	30 E1 1.09	31 F5 1.28	32 E3 0.97	33 E5 0.73 BK5
34 E0 0.49	35 F1 1.16	36 D 0.97	37 F4 1.27	38 E1 1.09	39 E3 1.03	40 E2 1.05	41 F5 1.26	42 B 0.85
43 E0 0.68	44 E0 1.16	45 F5 1.18 PL	46 D 0.88	47 F5 1.28	48 E2 1.05	49 F4 1.23	50 E4 1.05	51 F3 1.34
52 E4 0.69	53 F1 1.33	54 D 0.92	55 F5 1.21	56 E3 0.97	57 F5 1.26	58 E4 1.05	59 D 0.89	60 E5 1.10
61 F0 0.91	62 C 0.94	63 F4 1.22	64 E1 0.97	65 E5 0.73 BK5	66 B 0.85	67 F3 1.34	68 E5 1.10	69 B 0.78 PL

NOTE: X = MAXIMUM F_{xy} = 1.54

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES BOC, BANK 5 & PLCEA'S, HFP, ARO EQ. XENON	FIGURE 5-8

AA BB CC PL	AA = Quarter Core Location BB = Bat.. Type CC = R... ive Power Density PL = Part Length CEA Location

					1 E0 0.45	2 E0 0.59	3 E4 0.60	4 F0 0.77
			5 D 0.37	6 E1 0.61	7 F1 0.98	8 E0 0.93	9 F1 1.13	10 C 0.84
		11 E2 0.42	12 E6 0.76	13 F1 1.10	14 D 0.87	15 F5 1.15 PL	16 D 0.92	17 F4 1.38
	18 D 0.37	19 E6 0.76	20 F3 1.17	21 E4 1.02	22 F4 1.36	23 D 0.95	24 F5 1.42	25 E1 1.13
	26 E1 0.61	27 F1 1.10	28 E4 1.02	29 F2 1.18 PL	30 E1 1.07	31 F5 1.45	32 E3 1.13	33 E5 1.13
34 E0 0.45	35 F1 0.97	36 D 0.87	37 F4 1.36	38 E1 1.07	39 E3 1.05	40 E2 1.14	41 F5 1.51	42 B 1.03
43 E0 0.59	44 E0 0.93	45 F5 1.15 PL	46 D 0.95	47 F5 1.45	48 E2 1.14	49 F4 1.51	50 E4 1.13	51 F3 1.41
52 E4 0.60	53 F1 1.13	54 D 0.92	55 F5 1.42	56 E3 1.13	57 F5 1.51 X	58 E4 1.13	59 D 0.91	60 E5 1.04
61 F0 0.77	62 C 0.85	63 F4 1.38	64 E1 1.13	65 E5 1.13	66 B 1.03	67 F3 1.41	68 E5 1.04	69 B 0.74 PL

NOTE: X = MAXIMUM F_{xy} = 1.59

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES EOC, PLCEA'S INSERTED, HFP, ARO EQ. XENON	FIGURE 5-9

AA BB CC BK5

AA = Quarter Core Location
BB = Batch Type
CC = Relative Power Density
BK5 = CEA Bank 5 Location

					1 E0 0.49	2 E0 0.64	3 E4 0.63	4 F0 0.81
			5 D 0.41	6 E1 0.67	7 F1 1.08	8 E0 1.03	9 F1 1.20	10 C 0.87
		11 E2 0.47	12 E6 0.84	13 F1 1.22	14 D 0.97	15 F5 1.39	16 D 0.95	17 F4 1.35
	18 D 0.41	19 E6 0.84	20 F3 1.30	21 E4 1.14	22 F4 1.47 X	23 D 0.97	24 F5 1.31	25 E1 0.97
	26 E1 0.67	27 F1 1.22	28 E4 1.14	29 F2 1.45	30 E1 1.12	31 F5 1.38	32 E3 0.94	33 E5 0.66 BK5
34 E0 0.49	35 F1 1.08	36 D 0.97	37 F4 1.47	38 E1 1.12	39 E3 1.02	40 E2 1.03	41 F5 1.27	42 B 0.81
43 E0 0.64	44 E0 1.03	45 F5 1.39	46 D 0.97	47 F5 1.38	48 E2 1.03	49 F4 1.33	50 E4 0.97	51 F3 1.19
52 E4 0.63	53 F1 1.20	54 D 0.95	55 F5 1.31	56 E3 0.94	57 F5 1.27	58 E4 0.96	59 D 0.79	60 E5 0.93
61 F0 0.81	62 C 0.88	63 F4 1.35	64 E1 0.97	65 E5 0.67 BK5	66 B 0.81	67 F3 1.19	68 E5 0.92	69 B 0.74

NOTE: X = MAXIMUM F_{xy} = 1.56

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIES EOC, BANK 5 INSERTED, HFP, ARO EQ. XENON	FIGURE 5-10

AA BB CC BK5/PL

AA = Quarter Core Location
BB = Batch Type
CC = Relative Power Density
PL = Part Length CEA Location
BK5 = Bank 5 CEA Location

					1 E0 0.48	2 E0 0.64	3 E4 0.63	4 F0 0.82
			5 D 0.41	6 E1 0.66	7 F1 1.05	8 E0 0.99	9 F1 1.18	10 C 0.87
		11 E2 0.47	12 E6 0.84	13 F1 1.20	14 D 0.93	15 F5 1.19 PL	16 D 0.93	17 F4 1.36
	18 D 0.41	19 E6 0.84	20 F3 1.27	21 E4 1.09	22 F4 1.42	23 D 0.95	24 F5 1.32	25 E1 0.99
	26 E1 0.66	27 F1 1.20	28 E4 1.09	29 F2 1.24 PL	30 E1 1.10	31 F5 1.41	32 E3 0.99	33 E5 0.71 BK5
34 E0 0.48	35 F1 1.05	36 D 0.93	37 F4 1.42	38 E1 1.10	39 E3 1.05	40 E2 1.10	41 F5 1.38	42 B 0.89
43 E0 0.64	44 E0 0.99	45 F5 1.19 PL	46 D 0.95	47 F5 1.41	48 E2 1.10	49 F4 1.45 X	50 E4 1.07	51 F3 1.32
52 E4 0.64	53 F1 1.18	54 D 0.93	55 F5 1.32	56 E3 0.99	57 F5 1.38	58 E4 1.06	59 D 0.87	60 E5 1.00
61 F0 0.82	62 C 0.88	63 F4 1.36	64 E1 0.99	65 E5 0.71 BK5	66 B 0.89	67 F3 1.32	68 E5 1.00	69 B 0.71 PL

NOTE: X = MAXIMUM F_{xy} = 1.52

PALO VERDE NUCLEAR GENERATING STATION Unit 1	PALO VERDE UNIT 1 CYCLE 4 ASSEMBLY RELATIVE POWER DENSITIFS EOC, BANK 5 & PLCEA'S, HFP, ARO EQ. XENON	FIGURE 5-11

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR ANALYSIS

Steady state DNBR analyses of Cycle 4 at * rated power level of 3800 MWT have been performed using the TORC compu*er code described in Reference 6-1, the CE-1 critical heat flux correlation described in References 6-2 and 6-8, and the CETOP code described in Reference 6-3.

Table 6-1 contains a list of pertinent thermal-hydraulic design parameters. The Modified Statistical Combination of Uncertainties (MSCU) methodology presented in Reference 6-4 was applied with Palo Verde-1 specific data using the calculational factors listed in Table 6-1 and other uncertainty factors to define overall uncertainty penalty factors to be applied in the DNBR calculations performed by the Core Protection Calculators (CPC) and Core Operating Limit Supervisory System (COLSS) which, when used with the Cycle 4 DNBR limit of 1.24, provide assurance at the 95/95 confidence/probability level that the hot rod will not experience DNB. The 1.24 DNBR limit was calculated using the methodology of Reference 6-5 as was done for the Reference Cycle.

This Cycle 4 DNBR limit includes the following allowances:

1. NRC imposed 0.01 DNBR penalty for HID-1 grids as discussed in Reference 6-6.

2. Rod bow penalty as discussed in Section 6.2 below.

Other penalties imposed by NRC in the course of their review of the Cycle 1 Statistical Combination of Uncertainties (SCU) analysis discussed in Reference 6-5 (i.e., TORC code uncertainty and CE-1 CHF correlation cross validation uncertainty, as discussed in Reference 6-6) are included in the overall uncertainty penalty factors derived in the Cycle 4 MSCU analysis.

6.2 EFFECTS OF FUEL ROD BOWING ON DNBR MARGIN

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in Reference 6-7. The penalty used for this analysis, 1.75% MDNBR, is valid for bundle burnups up to 30 GWD/T. This penalty is included in the 1.24 DNBR limit.

For assemblies with burnup greater than 30 GWD/T sufficient available margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Hence the rod bow penalty based upon Reference 6-7 for 30 GWD/T is applicable for all assembly burnups expected for Cycle 4.

Table 6-1

PVNGS-1 Cycle 4
Thermal Hydraulic Parameters at Full Power

General Characteristics	Units	Reference Cycle	PVNGS-1 Cycle 4
Total Heat Output (Core only)	MWt E6 Btu/hr	3800 12,970	3800 12,970
Fraction of Heat Generated in Fuel Rod	- - -	0.975	0.975
Primary System Pressure (Nominal)	psia	2250	2250
Inlet Temperature (Nominal)	°F	565.0	565.0
Total Reactor Coolant Flow ++ (Minimum steady state)	gpm E6 lbm/hr	423,300 155.8	423,300 155.8
Coolant Flow Through Core (Minimum)	E6 lbm/hr	151.1	151.1
Hydraulic Diameter (Nominal channel)	ft	0.039	0.039
Average Mass Velocity	E6 lbm/hr-ft	2.49	2.49
Minimum Pressure Drop Across Core Steady State Flow Irreversible ΔP Over Entire Fuel Assembly	psid	14.5	14.5
Total Pressure Across Vessel (Based on nominal dimensions and minimum steady state flow)	psid	51.3	51.3
Core Average Heat Flux (Accounts for fraction of heat generated in fuel rod and axial densification factor)	Btu/hr-ft^2	184,200*	185,300**
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	68,600*	68,200**

Table 6-1 (continued)

General Characteristics	Units	Reference Cycle	PVNGS-1 Cycle 4
Film Coefficient at Average Conditions	Btu/hr-ft^2-°F	6100	6100
Average Film Temperature Difference	°F	30	30
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod)	kW/ft	5.4	5.4
Average Core Enthalpy Rise	Btu/lbm	85.9	85.9
Maximum Clad Surface Temperature	°F	656	656
Engineering Heat Flux Factor	- - -	1.03 +	1.03 +
Engineering Factor on Hot Channel Heat Input	- - -	1.03 +	1.03 +
Rod Pitch, Bowing and Clad Diameter Factor	- - -	1.05 +	1.05 +
Fuel Densification Factor (Axial)	- - -	1.002	1.002

Notes:

* Based on 1872 poison rods.

** Based on 2176 poison rods.

+ These factors have been combined statistically with other uncertainty factors as described in Reference 6-4 to define overall uncertainty adjustment factors to be applied in the DNBR calculations in COLSS and CPC which, when used in conjunction with the DNBR limit provides assurance at the 95 / 95 confidence / probability level that the hot rod will not experience DNB.

++ Technical Specification minimum flowrate.

7.0 NON-LOCA TRANSIENT ANALYSIS

7.0.1 Introduction

This section presents the results of the Palo Verde Nuclear Generating Station Unit 1 (PVNGS-1), Cycle 4 Non-LOCA safety analyses at 3800 MWt.

The Design Basis Events (DBEs) considered in the safety analyses are listed in Table 7.0-1. These events are categorized into three groups: Moderate Frequency, Infrequent, and Limiting Fault events. For the purpose of this report, the Moderate Frequency and Infrequent Events will be termed Anticipated Operational Occurrences. The DBEs were evaluated with respect to four criteria: Offsite Dose, Reactor Coolant System (RCS) Pressure, Fuel Performance (DNBR and Centerline Melt SAFDLs), and Loss of Shutdown Margin. Tables 7.0-2 through 7.0-5 present the lists of events analyzed for each criterion. All events were re-evaluated to assure that they meet their respective criteria for Cycle 4. The DBEs chosen for analysis for each criterion are the limiting events with respect to that criterion.

7.0.2 Methods of Analysis

The analytical methodology used for PVNGS-1 Cycle 4 is the same as the Unit 1 Cycle 3 (Reference Cycle) methodology (References 7-1, 7-2 and 7-9) with the exception of event 7.1.4, the Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve with a Loss of Offsite Power for which Unit 3 Cycle 3 (Reference 7-11) forms the Reference Cycle as it represents the latest NRC position on the analysis of this event. Only methodology that has previously been reviewed and approved on the PVNGS dockets (References 7-9, 7-10 and 7-11), and/or the CESSAR docket (Reference 7-2) is used.

7.0.3 Mathematical Models

The mathematical models and computer codes used in the Cycle 4 Non-LOCA safety analysis are the same as those used in the Reference Cycle analysis (References 7-1, 7-2 and 7-9). Plant response for Non-LOCA Events was simulated using the CESEC III computer code (Reference 7-3). Simulation of the fluid conditions within the hot channel of the reactor core and calculation of DNBR was performed using the CETOP-D computer code that was verified to be applicable in Reference 7-4.

The TORC computer code was used to simulate the fluid conditions within the reactor core and to calculate fuel pin DNBR for the RCP Shaft Seizure and Sheared Shaft event. The TORC code is described in References 7-6 and 7-7.

The number of fuel pins predicted to experience clad failure is taken as the number of pins which have a CE-1 DNBR value below 1.24. The exceptions are the CEA Ejection, the Shaft Seizure, Sheared Shaft and the Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve with a Loss of Offsite Power events for which the statistical convolution method, described in Reference 7-8, was used. Reference 7-8 has been approved by the NRC and has been used in References 7-1, 7-2, 7-9, 7-10 and 7-11.

The HERMITE computer code (Reference 7-5) was used to simulate the reactor core for analyses which required more spatial detail than is provided by a point kinetics model. Reference 7-5 has been approved by the NRC and has been used in References 7-1, 7-2 and 7-9. HERMITE was also used to generate input to the CESEC point kinetics model by partially crediting space-time effects so that the CESEC calculation of core power during a reactor scram is conservative relative to HERMITE.

7.0.4 Input Parameters and Analysis Assumptions

Table 7.0-6 summarizes the core parameters assumed in the Cycle 4 transient analysis and compares them to the values used in the Reference Cycle. Specific initial conditions for each event are tabulated in the section of the report summarizing that event. Changes in the Technical Specifications that are necessary for the operation of Cycle 4 are described in Section 10. The effects of these changes were considered for each DBE and were included as appropriate. For some of the DBEs presented, certain initial core parameters were assumed to be more limiting than the actual calculated Cycle 4 values. Such assumptions resulted in more adverse consequences. Events which have credited CPC trip protection have assumed instrument channel response times which are conservative relative to the Unit 1 Technical Specifications.

7.0.5 Conclusion

All DBEs have been evaluated for PVNGS-1, Cycle 4 to determine whether their results are bounded by the Reference Cycle.

Table 7.0-1

PVNGS Unit 1 Design Basis
Events Considered in the Cycle 4 Safety Analysis

7.1 Increase in Heat Removal by the Secondary System

- 7.1.1 Decrease in Feedwater Temperature
- 7.1.2 Increase in Feedwater Flow
- 7.1.3 Increased Main Steam Flow
- 7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve
- 7.1.5* Steam System Piping Failures

7.2 Decrease in Heat Removal by the Secondary System

- 7.2.1 Loss of External Load
- 7.2.2 Turbine Trip
- 7.2.3 Loss of Condenser Vacuum
- 7.2.4 Loss of Normal AC Power
- 7.2.5 Loss of Normal Feedwater
- 7.2.6* Feedwater System Pipe Breaks

7.3 Decrease in Reactor Coolant Flowrate

- 7.3.1 Total Loss of Forced Reactor Coolant Flow
- 7.3.2* Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

7.4 Reactivity and Power Distribution Anomalies

- 7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition
- 7.4.2 Uncontrolled CEA Withdrawal at Power
- 7.4.3 CEA Misoperation Events
- 7.4.4 CVCS Malfunction (Inadvertent Boron Dilution)
- 7.4.5 Startup of an Inactive Reactor Coolant Pump
- 7.4.6* Control Element Assembly Ejection

7.5 Increase in Reactor Coolant System Inventory

- 7.5.1 CVCS Malfunction
- 7.5.2 Inadvertent Operation of the ECCS During Power Operation

* Categorized as Limiting Fault Events

Table 7.0-1 (continued)

7.6 Decrease in Reactor Coolant System Inventory

- 7.6.1 Pressurizer Pressure Decrease Events
- 7.6.2* Small Primary Line Break Outside Containment
- 7.6.3* Steam Generator Tube Rupture

7.7 Miscellaneous

- 7.7.1 Asymmetric Steam Generator Events

* Categorized as Limiting Fault Events

Table 7.0-2

DBEs Evaluated with Respect to Offsite Dose Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.2.4	2) Loss of Normal AC Power	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5a	a) Pre-Trip Power Excursions	
7.1.5b	b) Post-Trip Return-to-Power	
7.2.6	2) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.3.2	3) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Presented
7.4.5	4) Control Element Assembly Ejection	Bounded by Reference Cycle
7.6.2	5) Small Primary Line Break Outside Containment	Bounded by Reference Cycle
7.6.3	6) Steam Generator Tube Rupture	Bounded by Reference Cycle

Table 7.0-3

DBEs Evaluated with Respect to RCS Pressure Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.2.1	1) Loss of External Load	Bounded by Reference Cycle
7.2.2	2) Turbine Trip	Bounded by Reference Cycle
7.2.3	3) Loss of Condenser Vacuum	Bounded by Reference Cycle
7.2.4	4) Loss of Normal AC Power	Bounded by Reference Cycle
7.2.5	5) Loss of Normal Feedwater	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from Subcritical or Low Power Condition	Bounded by Reference Cycle
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.5.1	8) CVCS Malfunction	Bounded by Reference Cycle
7.5.2	9) Inadvertent Operation of the ECCS During Power Operation	Bounded by Reference Cycle
	B) Limiting Fault Events	
7.2.6	1) Feedwater System Pipe Breaks	Bounded by Reference Cycle
7.4.6	2) Control Element Assembly Ejection	Bounded by Reference Cycle

Table 7.0-4

DBEs Evaluated with Respect to Fuel Performance

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.1	1) Decrease in Feedwater Temperature	Bounded by Reference Cycle
7.1.2	2) Increase in Feedwater flow	Bounded by Reference Cycle
7.1.3	3) Increased Main Steam Flow	Bounded by Reference Cycle
7.1.4	4) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.3.1	5) Total Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.4.1	6) Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition	Bounded by Reference Cycle
7.4.2	7) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.4.3	8) CEA Misoperation Events	Bounded by Reference Cycle
7.6.1	9) Pressurizer Pressure Decrease Events	Bounded by Reference Cycle
7.7.1	10) Asymmetric Steam Generator Events	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5a	a) Pre-Trip Power Excursions	
7.1.5b	b) Post-Trip Return to Power	

Table 7.0-4 (continued)

Section	Event	Results
7.3.2	2) Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Presented
7.4.6	3) Control Element Assembly Ejection	Bounded by Reference Cycle

Table 7.0-5

DBEs Evaluated with Respect to Shutdown Margin Criterion

Section	Event	Results
	A) Anticipated Operational Occurrences	
7.1.4	1) Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Bounded by Reference Cycle
7.4.4	2) CVCS Malfunction (Inadvertent Boron Dilution)	Bounded by Reference Cycle
7.4.5	3) Startup of an Inactive Reactor Coolant System Pump	Bounded by Reference Cycle
	B) Limiting Fault Events	
	1) Steam System Piping Failures:	Bounded by Reference Cycle
7.1.5b	a) Post-Trip Return-to-Power	

Table 7.0-6

PVNGS Unit 1, Cycle 4
Core Parameters Input to Safety Analyses

Safety Parameters	Units	Reference Cycle Value	Cycle 4 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3898	3898
Core Inlet Steady State Temperature	°F	560 to 570 (90% power and above) 550 to 572 (below 90% power)	560 to 570 (90% power and above) 550 to 572 (below 90% power)
Steady State RCS Pressure	psia	2000 - 2325	2000 - 2325
Minimum Guaranteed Delivered Volumetric Flow Rate	gpm	423,320	423,320
Axial Shape Index LCO Band Assumed	ASI Units	-0.3 to +0.3 (≥ 20% Power) -0.6 to +0.6 (< 20% Power)	-0.3 to +0.3 (≥ 20% Power) -0.6 to +0.6 (< 20% Power)
Maximum CEA Insertion at Full Power	% Insertion of Lead Bank	28	28
	% Insertion of Part-Length	25	25
Maximum Initial Linear Heat Rate	KW/ft	13.5	13.5
Steady State Linear Heat Rate for Fuel Center Line Melt	KW/ft	21.0	21.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	4.0	4.0

Table 7.0-6 (continued)

Safety Parameters	Units	Reference Cycle Value	Cycle 4 Value
Minimum DNBR			
CE-1 (SAFDL)		1.24	1.24
MacBeth (Fuel failure limit for post-trip SLB with LOAC - References 7-12 and 7-13)		1.30	1.30
Initial Moderator Temperature Coefficient	10^{-4} $\Delta\rho$/°F	Figure 7.0-1	Figure 7.0-1
Shutdown Margin (Value Assumed in Limiting Hot Zero Power SLB)	%$\Delta\rho$	-6.5	-6.5

7.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

7.1.1 Decrease in Feedwater Temperature

The results are bounded by the Reference Cycle.

7.1.2 Increase in Feedwater Flow

The results are bounded by the Reference Cycle.

7.1.3 Increased Main Steam Flow

The results are bounded by the Reference Cycle.

7.1.4 Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve

The results are bounded by the Reference Cycle.

7.1.5 Steam System Piping Failures

7.1.5a Steam System Piping Failures: Inside and Outside Containment Pre-Trip Power Excursions

The results are bounded by the Reference Cycle.

7.1.5b Steam System Piping Failures: Post-Trip Return to Power

The results are bounded by the Reference Cycle

7.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

7.2.1 Loss of External Load

The results are bounded by the Reference Cycle.

7.2.2 Turbine Trip

The results are bounded by the Reference Cycle.

7.2.3 Loss of Condenser Vacuum

The results are bounded by the Reference Cycle.

7.2.4 Loss of Normal AC Power

The results are bounded by the Reference Cycle.

7.2.5 Loss of Normal Feedwater

The results are bounded by the Reference Cycle.

7.2.6 Feedwater System Pipe Breaks

The results are bounded by the Reference Cycle.

7.3 DECREASE IN REACTOR COOLANT FLOWRATE

7.3.1 Loss of Forced Reactor Coolant Flow

The results are bounded by the Reference Cycle.

7.3.2 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The amount of predicted failed fuel has increased for the Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft from 3.79 to 4.32 %, which is less than the 4.5 % predicted fuel failure found acceptable by the NRC in Reference 7-11. The increase in failed fuel was the result of more adverse nuclear power distributions. The resultant radiological consequences are a 2 hour site boundary thyroid dose of less than 240 Rem This is within 10CFR100 guidelines.

7.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

7.4.1 Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition

The results are bounded by the Reference Cycle.

7.4.2 Uncontrolled CEA Withdrawal at Power

The results are bounded by the Reference Cycle.

7.4.3 CEA Misoperation Event

The results are bounded by the Reference Cycle.

7.4.4 CVCS Malfunction (Inadvertent Boron Dilution)

The results are bounded by the Reference Cycle.

7.4.5 Startup of an Inactive Reactor Coolant Pump

The results are bounded by the Reference Cycle.

7.4.6 Control Element Assembly Ejection

The results are bounded by the Reference Cycle.

7.5 INCREASE IN REACTOR COOLANT SYSTEM INVENTORY

7.5.1 CVCS Malfunction

The results are bounded by the Reference Cycle.

7.5.2 Inadvertent Operation of the ECCS During Power Operation

The results are bounded by the Reference Cycle.

7.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY

7.6.1 Pressurizer Pressure Decrease Events

The results are bounded by the Reference Cycle.

7.6.2 Small Primary Line Break Outside Containment

The results are bounded by the Reference Cycle.

7.6.3 Steam Generator Tube Rupture

The results are bounded by the Reference Cycle.

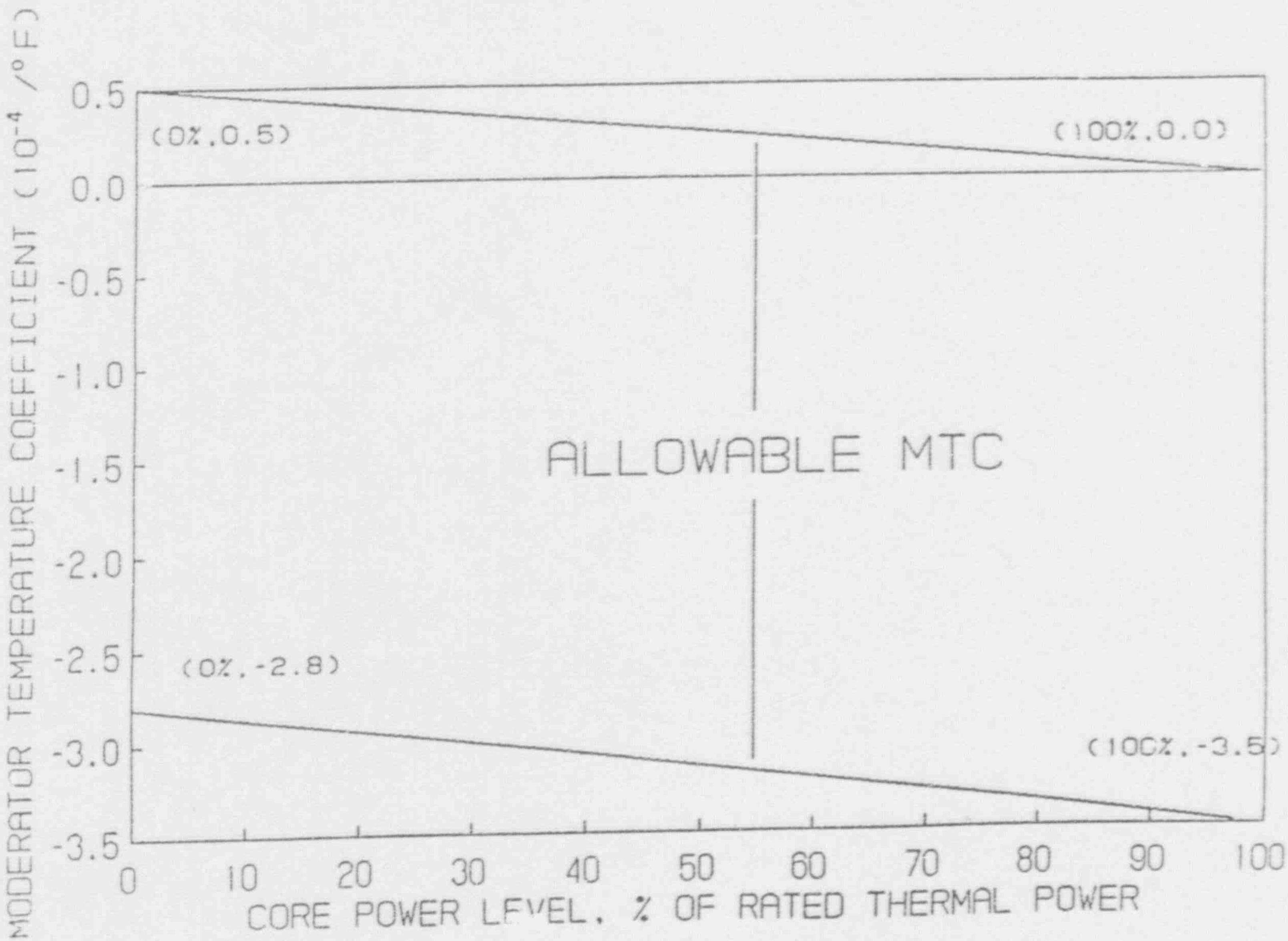
7.7 MISCELLANEOUS

7.7.1 Asymmetric Steam Generator Events

The results are bounded by the Reference Cycle.

FIGURE 7.0-1

ALLOWABLE MTC MODES 1 AND 2



8.0 ECCS ANALYSIS

8.1 LARGE BREAK LOSS-OF-COOLANT ACCIDENT

8.1.1 Introduction And Summary

An ECCS performance analysis of the limiting break size was performed for PVNGS-1 Cycle 4 to demonstrate compliance with 10CFR50.46 which presents the NRC Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled reactors (Reference 8-1). The analysis justifies an allowable Peak Linear Heat Generation Rate (PLHGR) of 13.5 kW/ft. The method of analysis and detailed results which support this value are presented herein.

8.1.2 Method Of Analysis

The large break ECCS performance analysis for PVNGS-1 Cycle 4 consisted of three parts: 1) an evaluation of the differences between Cycle 4 and Cycle 3, 2) a calculation of cladding temperature and oxidation for the hot rod for Cycle 4, and 3) a comparison of the results of the calculation to the results of PVNGS-1 Cycle 1. For this reason PVNGS-1, Cycle 1 which was the Reference Cycle for Cycle 3 is referred to as the Reference Cycle in Section 8 for Cycle 4. Acceptable ECCS performance was demonstrated for the Reference Cycle in Reference 8-2 and approved by the NRC in Reference 8-3. As in the Reference Cycle, the calculations performed for this evaluation used the NRC approved C-E large break ECCS performance evaluation model which is described in Reference 8-4 including the use of a more conservative axial power shape. The blowdown hydraulic calculations, refill/reflood hydraulics calculations, and steam cooling heat transfer coefficients of the Reference Cycle apply to PVNGS-1 Cycle 4 since there have been no significant adverse changes to RCS or ECCS hardware characteristics, or to core and system parameters introduced by Cycle 4. Therefore, only fuel rod cladding temperature and oxidation calculations are required to re-evaluate ECCS performance with respect to the changes in fuel conditions introduced by Cycle 4. The NRC approved STRIKIN-II (Reference 8-5) code was used for this purpose.

Burnup dependent calculations were performed with STRIKIN-II to determine the limiting conditions for the ECCS performance analysis. The fuel performance data were generated with the FATES-3A fuel evaluation model (References 8-6 and 8-7). It was demonstrated that the burnup with the highest initial fuel stored energy was limiting. This occurred at a low burnup for the hot rod.

The temperature and oxidation calculations were performed for the 1.0 Double-Ended Guillotine at Pump Discharge (DEG/PD) break. This break size is the limiting break size of the Reference Cycle and, as there are no significant differences between Cycle 4 and Cycle 1 that impact the hydraulic calculation, is the limiting break size for Cycle 4.

8.1.3 Results

The ECCS performance analysis for PVNGS-1 Cycle 4 showed that the Reference Analysis results conservatively apply. The peak cladding temperature, maximum local cladding oxidation, and core wide oxidation values of 2091°F, 9.0% and < 0.80%, respectively, for the Reference Analysis are below the corresponding 10CFR50.46 acceptance criteria of 2200°F, 17%, and 1%, respectively. These results remain applicable for up to 400 tubes plugged per steam generator and a reduction in system flow rate to 155.8×10^6 lbm/hr and a reduction in core flow rate to 151.1×10^6 lbm/hr.

8.1.4 Conclusion

Conformance to the ECCS criteria is demonstrated by the analysis results. Therefore, operation of PVNGS-1 Cycle 4 at a core power level of 3876 MWt (102% of 3800 MWt) and a PLHGR of 13.5 kW/ft is in compliance with 10CFR50.46.

8.2 SMALL BREAK LOSS-OF-COOLANT ACCIDENT

The small break ECCS performance analysis for PVNGS-1 Cycle 4 consisted of an evaluation of the differences between Cycle 4 and Cycle 3 and a comparison to the reported small break loss-of-coolant accident (LOCA) results (Reference 8-9) for PVNGS-1 Cycle 1. The analysis confirmed that the peak cladding temperature for the limiting small break LOCA remains more than 300 F below that of the limiting large break LOCA. Therefore, acceptable small break ECCS performance is demonstrated at a peak linear heat generation rate of 13.5 kW/ft and a reactor power level of 3876 MWt (102% of 3800 MWt). The acceptable performance has been confirmed with up to 400 plugged tubes per steam generator.

9.0 REACTOR PROTECTION AND MONITORING SYSTEM

9.1 INTRODUCTION

The Core Protection Calculator System (CPCS) is designed to provide the low DNBR and high Local Power Density (LPD) trips to (1) ensure that the specified acceptable fuel design limits on departure from nucleate boiling and centerline fuel melting are not exceeded during Anticipated Operational Occurrences (AOOs) and (2) assist the Engineered Safety Features System in limiting the consequences of certain postulated accidents.

The CPCS in conjunction with the remaining Reactor Protection System (RPS) must be capable of providing protection for certain specified design basis events, provided that at the initiation of these occurrences the Nuclear Steam Supply System, its subsystems, components and parameters are maintained within operating limits and Limiting Conditions for Operation (LCOs).

9.2 CPCS SOFTWARE MODIFICATIONS

The algorithms associated with the CPC Improvement Program (References 9-1, 9-2 and 9-3) which were implemented in Cycle 2, are applicable to this cycle. The values for the Reload Data Block constants will be evaluated for applicability consistent with the cycle design, performance and safety analyses. Any necessary change to the RDB constants will be installed in accordance with Reference 9-4.

9.3 ADDRESSABLE CONSTANTS

Certain CPC constants are addressable so that they can be changed as required during operation. Addressable constants include (1) constants that are measured during startup (e.g., shape annealing matrix, boundary point power correlation coefficients, and adjustments for planar radial peaking factors), (2) uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations (BERR0 through BERR4), (3) trip setpoints and (4) miscellaneous items (e.g., penalty factor multipliers, CEAC penalty factor time delay, pre-trip setpoints, CEAC inoperable flag, calibration constants, etc.).

Trip setpoints, uncertainty factors and other addressable constants will be determined for this cycle consistent with the software and methodology established in the CPC Improvement Program and the cycle design, performance and safety analyses. As for the Reference Cycle, uncertainty factors will be determined using the modified statistical combination of uncertainties method (Reference 9-5).

9.4 DIGITAL MONITORING SYSTEM (COLSS)

The Core Operating Limit Supervisory System (COLSS), described in Reference 9-6, is a monitoring system that initiates alarms if the LCO's on DNBR, peak linear heat rate, axial shape index, core power, or core azimuthal tilt are exceeded. The COLSS data base and uncertainties will be updated, as required, to reflect the reload core design.

10.0 TECHNICAL SPECIFICATIONS

This section provides a summary of the proposed changes to the Technical Specification for PVNGS-1 Cycle 4. The following changes are referenced by their appropriate Section number. Detailed change pages for the Technical Specification are presented elsewhere.

Section 3.2.4 and 4.2.4 (DNBR Margin):

Revise Figure 3.2-2, COLSS out of service DNBR limit line - CEACs operable and Figure 3.2-2A, COLSS out of service DNBR limit line - CEACs inoperable to reflect Cycle 4 core characteristics.

11.0 STARTUP TESTING

The planned startup test program associated with core performance is outlined below. The described tests verify that core performance is consistent with the engineering design and safety analysis. The program conforms to Reference 11-1, "Startup Test Programs", and supplements normal surveillance tests which are required by Technical Specifications (i.e., CEA drop time testing, RCS flow measurement, MTC verification, etc).

11.1. LOW POWER PHYSICS TESTS

11.1.1 Initial Criticality

Before initial criticality, the critical boron concentration (CBC) will be estimated for the essentially all rods out (EARO) condition. Initial criticality will be achieved by one of two methods. By the first method, all CEA groups would be fully withdrawn (with the exception of the lead regulating group which would be positioned at approximately mid-core) before the beginning of boron dilution. The boron concentration of the reactor coolant would then be reduced until criticality is attained. By the second method, the boron concentration would be adjusted to the EARO estimated CBC before any CEAs are withdrawn. Then the PLCEA, shutdown, and regulating CEA groups would be withdrawn in sequence to achieve criticality.

11.1.2 Critical Boron Concentration (CBC)

The CBC will be determined for the unrodded configuration and for a partially rodded configuration. The measured CBC values will be verified to be within $\pm 1\%$ $\Delta k/k$ of the predicted values.

11.1.3 Temperature Reactivity Coefficient

The isothermal temperature coefficient (ITC) will be measured at the Essentially All Rods Out (EARO) configuration. The coolant temperature will be varied and the resulting reactivity change will be measured. The measured values will be verified to be within $\pm 0.3 \times 10^{-4}$ $\Delta k/k/^{\circ}F$ of the predicted values.

11.1.4 CEA Reactivity Worth

CEA group worths will be measured using the CEA Exchange technique. This technique consists of measuring the worth of a "Reference Group" via standard boration/dilution techniques and then exchanging this group with other groups to measure their worths. All full-length CEAs will be included in the measurement. Due to the large differences in CEA group worths, two reference groups (one with high worth and one with medium worth) may be used. The groups to be measured will be exchanged with the appropriate reference group. Acceptance criteria will be as specified in Reference 11-2.

11.1.5 Inverse Boron Worth (IBW)

The IBW will be calculated using results from the CBC measurements and the CEA group worth measurements. The calculated IBW value will be verified to be within ± 15 ppm/% $\Delta k/k$ of the predicted value.

11.2 Power Ascension Testing

Following completion of the Low Power Physics Test sequence, reactor power will be increased in accordance with normal operating procedures. The power ascension will be monitored through use of an off-line NSSS performance and data processing computer algorithm. This computer code will be executed in parallel with the power ascension to monitor CPC and COLSS performance relative to the processed plant data against which they are normally calibrated. If necessary, the power ascension will be suspended while necessary data reduction and equipment calibrations are performed. The following measurements will be performed during the program.

11.2.1 Flux Symmetry Verification

Core power distribution, as determined from fixed incore detector data, will be examined prior to exceeding 30% power to verify that no detectable fuel misloadings exist. Differences between measured powers in symmetric, instrumented assemblies will be verified to be within 10% of the symmetric group average.

11.2.2 Core Power Distribution

Core power distributions derived from the fixed incore neutron detectors will be compared to predicted distributions at two power plateaus. These comparisons serve to further verify proper fuel loading and verify consistency between the as-built core and the engineering design models. Compliance with the acceptance criteria at the intermediate power plateau (between 40% and 70% power) provides reasonable assurance that the power distribution will remain within the design limits while reactor power is increased to 100%, where the second comparison will be performed.

The measured results will be compared to the predicted values in the following manner for both the intermediate and the full power analyses:

A. The root-mean-square (RMS) of the difference between the measured and predicted relative power density (axially integrated) for each of the fuel assemblies will be verified to be less than or equal to 5%.

B. The RMS of the difference between the measured and predicted core average axial power distribution for each axial node will be verified to be less than or equal to 5%.

C. The measured values of planar radial peaking factor (Fxy), integrated radial peaking factor (Fr), core average axial peak (F_z), and the 3-D power peak (Fq) will be verified to be within $\pm$10% of their predicted values.

11.2.3 Shape Annealing Matrix (SAM) and Boundary Point Power Correlation Coefficients (BPPCC) Verification

The SAM and BPPCC values will be determined from a linear regression analysis of the measured excore detector readings and corresponding core power distribution determined from incore detector signals. Since these values must be representative for a rodded and unrodded core throughout the cycle, it is desirable to use as wide a range of axial shapes as is available to establish their values. The spectrum of axial shapes encountered during the power ascension has been demonstrated to be adequate for the calculation of the matrix elements. The necessary data will be compiled and analyzed through the power ascension by the off-line NSSS performance and data processing algorithm. The results of the analysis will be used to modify the appropriate CPC constants, if necessary.

11.2.4 Radial Peaking Factor (RPF) and CEA Shadowing Factor (RSF) Verification

The RPF and RSF values will be determined using data collected from the fixed incore detectors and the excore detectors. Values will be determined for unrodded as well as rodded (lead regulating group and part-length group only) operating conditions. Appropriate CPC and/or COLSS constants will be modified based upon the calculated values. The rodded portions of this measurement may be deleted from the test program if appropriate adjustments are made to CPC and COLSS constants.

11.2.5 Temperature Reactivity Coefficients at Power

The moderator temperature coefficient (MTC) will be measured at power within 7 EFPD after the accumulation of 40 EFPD. The measured MTC will be obtained from a measured isothermal temperature coefficient (ITC) using a calculated fuel temperature coefficient (FTC). The ITC will be measured by changing coolant temperature, compensating with CEA motion, and maintaining power steady. The measured MTC will be compared to the MTC Technical Specification to verify compliance with the operating license. This comparison will be done in such a way as to account for the MTC measurement uncertainty.

11.2.6 Critical Boron Concentration

The CBC will be determined for conditions of full power, equilibrium xenon. The measured CBC will be verified to be within $\pm$50 ppm of the predicted value after adjustment for the bias observed between measured and predicted CBC values at zero power.

11.3 PROCEDURE IF ACCEPTANCE CRITERIA ARE NOT MET

The results of all tests will be reviewed by the plant's reactor engineering group. If the acceptance criteria of the startup physics tests are not met, an evaluation will be performed with assistance from the fuel vendor as needed.

12.0 REFERENCES

12.1 SECTION 1.0 REFERENCES

(1-1) "Palo Verde Nuclear Generating Station Unit No. 1, Final Safety Analysis Report," Arizona Public Service Company, Docket No. 50-528.

(1-2) "Palo Verde Nuclear Generating Station Unit 1 Reload Analysis Report for Unit 1 Cycle 3," 161-01620-DBK/BJA, January 18, 1989, 161-01858-DBK/SW, April, 19, 1989, 161-02064-WFC/KLMC, June 27, 1989, 161-02219-WFC/KLMC, August 25, 1989, 161-02294-WFC/RAB, September 11, 1989, 161-02812-WFC/ACR/KLMC, January 25, 1990.

12.2 SECTION 2.0 REFERENCES

None

12.3 SECTION 3.0 REFERENCES

None

12.4 SECTION 4.0 REFERENCES

(4-1) 161-02602-WFC/RAB/KLMC, "Palo Verde Nuclear Generating Station (PVNGS) Unit 1, Fuel Surveillance Test Results - Unit 1, End-of-Cycle 2," November 8, 1989.

(4-2) 161-01102-EEVB/PGN, "CEA Guide Tube Wear Inspection Results," June 9, 1988.

(4-3) CENPD-139-P-A, "C-E Fuel Evaluation Model," Combustion Engineering, Inc., July, 1974.

(4-4) CEN-161-(B)-P-A, "Improvements to Fuel Evaluation Model," August, 1989.

(4-5) Not Used

(4-6) 161-00453-JGH/SGB, "Fuel Assembly Guide Tube Wear Program for PVNGS Unit 2," August 20, 1987.

(4-7) "Palo Verde Nuclear Generating Station, Updated Final Safety Analysis Report," Section 4.2.4 - Testing and Inspection Plan, Arizona Public Service Company, (Unit 1 Docket No. 50-528).

(4-8) CESSAR SSER2, Section 4.2.5, "Evaluation Findings", September 1983.

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APPENDIX TO

RELOAD ANALYSIS REPORT FOR

PALO VERDE NUCLEAR GENERATING STATION UNIT 1 CYCLE 4

10 CFR 50.59 EVALUATION

Per the requirements of 10CFR 50.59, a licensee is allowed to make changes to the facility described in the safety analysis report without prior NRC approval provided that the proposed change does not involve either (1) a change in the plant Technical Specifications incorporated in the license or (2) an unreviewed safety question.

A change to the facility described in the safety analysis report involves an unreviewed safety question:

(i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased;

or

(ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created;

or

(iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This 10CFR 50.59 evaluation includes (1) a brief description of the change to the facility described in the safety analysis report, (2) a determination as to whether the change involves an unreviewed safety question and (3) a safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

(1) Description of Change

The original methods and computer codes used to analyze the nuclear design of the core are described in Chapter 4 of the Palo Verde Nuclear Generating Station Updated Safety Analysis Report which refers to the System 80 Combustion Engineering Standard Safety Analysis Report. As indicated above, a licensee is allowed to make changes to these methods and codes provided that the change does not involve either a Technical Specification change or an Unreviewed Safety Question.

The nuclear design methods and computer codes provide calculated values for the following nuclear design parameters:

- Reactivity
- Reactivity Coefficients
- Control Rod Worths
- Peaking Factors
- Power Distribution Related Factors

Several changes have been made to these methods and computer codes to (1) simplify their use, (2) improve their computational efficiency (e.g., the exchange of data between codes), and (3) enhance their calculational accuracy. Of the three types of changes, only the latter, enhancing their calculational accuracy, is most likely to significantly affect the numerical results. Since the results of nuclear design analysis are used as input to the transient safety analysis that considers accidents and malfunction of equipment important to safety, these changes must be evaluated to determine whether or not an unreviewed safety question is created.

The original methods and computer codes are described in C-E's proprietary Topical Report CENPD-266-P-A, "The ROCS & DIT Computer Codes for Nuclear Design," dated April 1983. This Topical Report was generically reviewed and approved by the NRC[1]. Subsequent to the NRC's approval, changes were made to the methods and codes that could affect the calculational accuracy of the nuclear design computer codes. These changes are as follows:

- Implementation of Nodal Expansion Method to ROCS
- Improved Accounting of Anisotropic Scattering and Higher Order Interface Current Angular Distributions in DIT
- Use of Assembly Discontinuity Factors between ROCS and DIT
- Update of Biases and Uncertainties Applied to Calculated Parameters

A description of each change is provided below. The descriptions provide sufficient detail to perform a safety evaluation. Extensive reference is made to C-E proprietary documents that contain additional details regarding the numerical effect of each change.

Nodal Expansion Method

The Nodal Expansion Method (NEM) was added to the ROCS code as an alternative to the original Higher Order Difference (HOD) formulation. The ROCS code provides reactor power distributions and effective neutron multiplication factors. This data is then used to derive control rod worths, depletion, reactivity coefficients and reactivity differentials. Use of the NEM achieves significant reduction in computer running times and also improves agreement with fuel management measurement data.[2]

Although the NEM had not yet been fully integrated into the ROCS code, the use of the NEM was fully described in C-E Topical Report CENPD-266 that was approved by the NRC. Specifically, Topical Report CENPD-266 explained that NEM had been incorporated into a version of C-E's coarse-mesh kinetics code, HERMITE. Furthermore, Topical Report CENPD-266 presented numerical comparisons of the NEM and HOD methods for solving the neutron diffusion equations. The results showed that the substitution of NEM for the HOD method in ROCS would not have a significant impact on calculational results and uncertainties.

In recognition of the expected future implementation of the NEM into ROCS, the NRC stated the following in the Safety Evaluation Report (SER) that approved C-E Topical Report CENPD-266:

> *"We have reviewed the ROCS and DIT computer codes as described in CENPD-266-P and CENPD-266-NP and find them to be acceptable for nuclear core design and safety-related neutronics calculations made by CE in licensing actions for power distributions, control rod worths, depletion, reactivity coefficients and reactivity differential. We also conclude that the ROCS code, including the fine-mesh module MC, is of sufficient accuracy for the generation of coefficient libraries for the in-core instrumentation.*
>
> *The staff, however, recommends that CE perform further verification when the NEM is incorporated into the ROCS code in order to be assured that equivalent calculational biases and uncertainties are obtained with ROCS-NEM as compared to ROCS-HOD."*

Before using ROCS-NEM for nuclear design analysis for Palo Verde, C-E performed extensive verification to confirm that the calculational biases and uncertainties obtained with ROCS-NEM are equivalent to ROCS-HOD. The SER did not require C-E to resubmit the ROCS-NEM version of the code to the NRC for approval. It is important to note, however, that the NRC did recommend that the biases and uncertainties obtained when NEM was incorporated into ROCS be equivalent when compared to ROCS-HOD. By equivalent, it is understood that the results between the two methods need not be numerically identical, but rather that the two methods be equal to the degree that the same conservative relationship is maintained between calculated and measured data (i.e., a 95/95 tolerance limit).

C-E has performed detailed studies[4,5,6] to confirm that the ROCS-NEM nuclear core design and safety-related neutronics calculations of power distributions, control rod worths, depletion, reactivity coefficients and reactivity differentials maintain the same conservative relationship between calculated and measured data. In particular, the tolerance limits applied to the calculated results from ROCS-HOD and ROCS-NEM are identically defined as "the value that must be added to the calculated results to assure that 95% of the calculated values will be greater than the "true" value with 95% confidence." Thus, the change which adds NEM to ROCS has been demonstrated to be equivalent to the ROCS-HOD version, which was approved by the NRC.

Anisotropic Scattering and Higher Order Interface Current Angular Distributions

In order to maintain the calculational accuracy in C-E Topical Report CENPD-266 when evaluating fuel containing gadolinium as a burnable poison, C-E had to improve the way the nuclear design computer code accounted for the effects of anisotropic scattering and higher order interface current angular distributions in the DIT code. The DIT code is a transport theory-based code which performs spectral and spatial calculations in fuel cell and fuel assembly geometries. The DIT calculations provide few group neutron cross sections for use by the ROCS code.

The improved method for accounting for anisotropic scattering and higher order interface current angular distributions was submitted by C-E in a generic Topical Report which was reviewed and approved by the NRC[7]. These approved methods and computer codes are described in C-E Topical Report CENPD-275-P Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," dated May 1988. Although these changes were motivated by the need to obtain additional calculational accuracy to analyze gadolinium as a burnable poison, the method itself is independent of the burnable absorber used in the core.

Topical Report CENPD-275 was not submitted a a plant specific docket. It was reviewed by the NRC for generic implementation on PWR cores. In recognition of the generic applicability of the improvements made to the DIT code, the NRC stated the following in the SER that approved C-E Topical Report CENPD-275:

> *"We have reviewed the changes made to the DIT and ROCS/MC codes and methodology to accommodate the use of the integral burnable absorber gadolinium in PWR cores. These changes are typical of the types made by the industry for computing gadolinia cores. The numerical results that were provided show that acceptable agreement has been obtained between detailed calculations and design calculations. We conclude therefore that the changes made to the DIT and ROCS/MC codes and methodology are acceptable."*

> *"We also conclude that the neutronics methods described in the report (DIT, ROCS/MC, and PDQ), as modified, are acceptable for calculating the neutronic characteristics of PWR cores containing up to 8 weight percent gadolinia bearing fuel rods."*

It is also important to note that benchmark studies[3] and benchmark analysis provided in Topical Report CENPD-275 validated the changes made in the DIT code with B_4C poison that contained no gadolinium. The NRC SER, thus, concluded that the methods described in Topical Report CENPD-275 are acceptable for calculating the neutronic characteristics of PWR cores containing up to 8 weight percent gadolinia bearing fuel rods. This includes the case where the PWR core contains <u>zero</u> weight percent gadolinia by virtue of the fact that many of the assemblies used for benchmarking purposes did not contain <u>any</u> gadolinium bearing fuel rods. Indeed, the NRC also noted in the SER the following:

> *"The results obtained for the Lead Test Assemblies (LTA) are consistent with those obtained for the non-gadolinium bearing fuel assemblies. The staff concurs with CE's conclusion that these results provide additional validation of the DIT code and methodology."*

<u>Assembly Discontinuity Factors</u>

Assembly discontinuity factors (ADFs) are used in the nuclear industry[8] as a method to eliminate homogenization error in nuclear design analysis where the global heterogenous solution is known. The use of ADFs improves the internal agreement between the DIT and ROCS codes. The ADFs are derived from the very assembly calculations required by the conventional homogenization methods and, therefore, they do not add any new information to the overall calculational methodology. Thus, the use of the ADFs is expected to improve the accuracy of results obtained from ROCS when compared to DIT. In several detailed studies[4,5,6], C-E has confirmed that the assembly discontinuity factors improve the accuracy of the nuclear design analysis method and computer codes.

Biases and Uncertainties

In view of the above changes that have been made to the methods and nuclear design computer codes, the biases and uncertainties applied to the nuclear design parameters were formally reevaluated by C-E[9]. For nuclear design parameters, the bias represents either the average of measurement minus calculation or the average ratio of the difference between measurement and calculation to the calculation. The uncertainty value represents the 95/95 tolerance range for the parameter of interest.

The reevaluation produced revised bias and uncertainty values that are equivalent to those reported in C-E Topical Report CENPD-266. By equivalent, it is meant that the results are not numerically identical, but rather that their application preserves the same conservative statistical relationship between calculated and measured data (i.e., the 95/95 probability/confidence level).

The methods used to generate the new biases and uncertainties are the same as that described in Topical Report CENPD-266, with the exceptions of the method used to determine the bias and uncertainty for the net (N-1) rod worth. In the Topical Report, the bias and uncertainty associated with net (N-1) rod worth were calculated by evaluating the net rod worth measurements performed during initial core startups. These evaluations found that the calculated (N-1) rod worth was being under-predicted by 3.6%, with a 1.47% standard deviation about the mean value.

This standard deviation is quite small and was, therefore, considered inappropriate for use in the reload analysis process for two reasons. First, the (N-1) statistics were based on only a small number of (N-1) rod worth measurements. Second, the (N-1) measurements were taken during the beginning of cycle for the initial cores and, hence, may not adequately represent reload cores. In view of the limitations of the (N-1) statistics available when the

Topical Report was written, C-E chose to take a conservative approach. Specifically, C-E applied the larger bias and uncertainty associated with individual bank worths to the (N-1) rod worth. The evaluation of bias and uncertainty for the individual bank worths exhibited a mean over-prediction of 4% with a standard deviation of 4.2%.

Using the uncertainty for an individual bank for the (N-1) rod worth is also overly conservative because the maximum individual rod uncertainty is often dominated by rod banks with low worths. For low worth rod banks, the percentage uncertainty is often high despite the fact that the absolute value of the uncertainty is small and well within the experimental precision.

When C-E reevaluated the bias and uncertainty for the (N-1) configuration, C-E used the bias and uncertainty associated with the sum of the bank worths (i.e., "total" worth) in lieu of that for individual banks. The use of the total rod worth uncertainty is considered more appropriate than the individual bank worth since the total rod worth configuration is more representative of the higher control rod density of the (N-1) configuration. For this case, the calculated and measured data exhibited a mean under-prediction of 4.32% with a standard deviation of 1.97%.

This change in the bias and uncertainty used for the (N-1) case remains conservative because actual (N-1) measurements demonstrate that the uncertainty of the (N-1) rod worth is lower than the uncertainty of the total worth. This is expected since the (N-1) configuration is strongly influenced by the reactivity of the unrodded region of the core. Thus, the (N-1) configuration is less sensitive to the precision of the calculated effective control rod cross sections than are either the total or individual bank configurations.

The change in method to calculate the (N-1) rod worth produces equivalent set of bias and uncertainty, wherein the same conservative relationship is maintained between calculated and measured data (i.e., a 95/95 tolerance limit).

A comparison between the (N-1) rod worth using the original bias and uncertainty described in the Topical Report and those for new method is provided in Table 1 for past Palo Verde reload cycles. It can be seen that the specific change in rod worth is not dramatic, and in some cases non-existent. The changes, however, do indicate that there is generally more scram worth available than previous calculations suggested. In all cases, a 95/95 tolerance limit is still maintained between the calculated and measured results.

Table 1

PVNGS Unit/Cycle	(N-1) Rod Worth [% delta rho] with Original Bias & Uncertainty	Revised Bias & Uncertainty
1/2	7.0	7.1
2/2	7.2	7.4
3/2	7.0	7.0
1/3	6.5	6.4
2/3	6.4	6.4

(2) Unreviewed Safety Question Determination

The changes to the nuclear design analysis methods and computer codes described above can be implemented without prior NRC approval since there are no required changes to the Technical Specifications and an unreviewed safety question does not exist.

(3) Safety Evaluation

The determination that the changes to the nuclear design analysis methods and computer codes described above do not create an unreviewed safety question is demonstrated by the following:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased by the changes to the nuclear design analysis methods and computer codes described above.

The results of nuclear design analyses are used as inputs to the analysis of accidents or malfunction of equipment important to safety that are evaluated in the safety analysis report. These inputs do not alter the physical characteristics of any component involved in the initiation of an accident or any subsequent equipment malfunction. Thus, there is no increase in the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report as a result of this change.

The consequences of an accident or malfunction of equipment important to safety evaluated in the safety analysis report is affected by the value of inputs to the transient safety analysis. There is always the potential for the value of the nuclear design parameters to change solely as a result of the new reload fuel core loading pattern. Regardless of the source of a change, an assessment is always made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents and equipment malfunctions previously evaluated in the safety analysis.

If increased consequences are anticipated, compensatory actions are implemented to neutralize any expected increase in consequences. These compensatory actions include, but are not limited to, crediting any existing margins in the analysis or redefining the operating envelope to avoid increase consequences. Thus, the nuclear design parameters are intermediate results and by themselves will not result in a increase in the consequence of accident or malfunction of equipment important to safety evaluated in the safety analysis report.

Therefore, the changes to the nuclear design analysis methods and computer codes described above do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created by the changes to the nuclear design analysis methods and computer codes described above.

As noted above, the results of nuclear design analysis are used as inputs to the transient safety analysis of accidents or malfunction of equipment important to safety that are evaluated in the safety analysis report. These inputs do not alter the physical characteristics of any component involved in the initiation of an accident or any subsequent equipment malfunction. Thus, there is no increase in the possibility of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report as a result of this change.

Thus, the changes to the nuclear design analysis methods and computer codes described above will not create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

3. The margin of safety as defined in the basis for any technical specification will not be reduced by the changes to the nuclear design analysis methods and computer codes described above.

Extensive benchmarking of the new nuclear design methods and computer codes has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained using the previous methods and computer codes. For any changes in the calculated values that do occur, the reevaluation of the biases and uncertainties ensures that the current margin of safety is maintained. Specifically, use of these revised biases and uncertainties in safety evaluations continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis do not exceed the actual values on at least a 95/95 probability/confidence basis.

The changes to the nuclear design analysis methods and computer codes described above, therefore, do not reduce the margin of safety as defined in the basis for any technical specification.

In conclusion, the changes to the nuclear design analysis methods and computer codes described above do not involve an unreviewed safety question and does not require a change to the Technical Specifications. Therefore, prior NRC approval is not required for this change.

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