ATTACHMENT

PVNGS

UNIT 2, CYCLE 3

RELOAD ANALYSIS REPORT

RELOAD ANALYSIS REPORT FOR
PALO VERDE NUCLEAR GENERATING STATION UNIT 2
CYCLE 3

TABLE OF CONTENTS

[illegible]

1.0 INTRODUCTION AND SUMMARY

This report provides an evaluation of the design and performance of Palo Verde Nuclear Generating Station Unit 2 (PVNGS-2) during its third cycle of operation at 100% rated core power of 3800 MWt and NSSS power of 3822 MWt. Operating conditions for Cycle 3 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 and D assemblies, along with fresh Batch E assemblies. The Cycle 2 termination burnup has been assumed to be between 394 and 446 EFPD (Effective Full Power Days).

The second 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 all postulated accidents and anticipated operational occurrences has shown that the Cycle 3 core design meets these safety criteria.

The Cycle 3 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 3 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 not bounded by those of the Reference Cycle analysis, therefore, reanalysis is required.

[illegible] [illegible]

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 3 are summarized in Section 10 and described in greater detail in separate license amendment applications.

[illegible]

2.0 OPERATING HISTORY OF THE REFERENCE CYCLE

The Reference Cycle began with initial criticality on May 15, 1988. Power ascension began on May 23, 1988, and on May 30, 1988, the unit reached full power.

It is presently estimated that Cycle 2 will terminate on or about February 14, 1990. The Cycle 2 termination point can vary between 394 and 446 EFPD to accommodate the plant schedule and still be within the assumptions of the Cycle 3 analyses.

[illegible]

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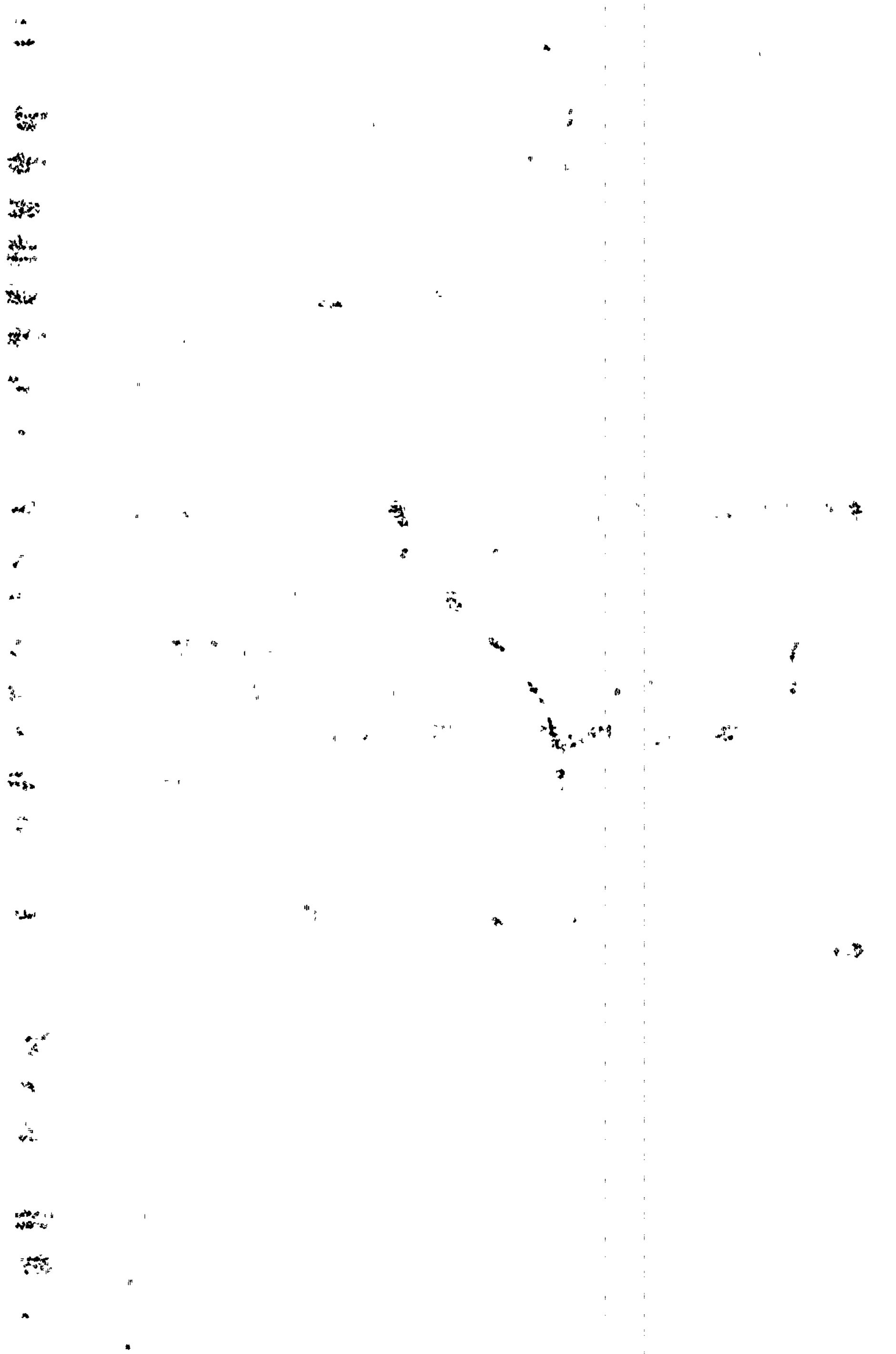


3.0 GENERAL DESCRIPTION

The Cycle 3 core will consist of those assembly types and numbers listed in Table 3-1. Sixty-nine Batch B assemblies and twenty-eight Batch C will be removed from the Cycle 2 core to make way for ninety-six fresh, Batch E assemblies. Thirty-six Batch C and all Batch D assemblies now in the core will be retained. One Batch B assembly discharged at EOC1 will be reinserted into the core. Figure 3-1 shows the poison shim and zoning configuration for those assemblies.

The reload batch will consist of 24 type E0 assemblies, 16 type E1 assemblies with 16 burnable poison shims per assembly, 24 type E2 assemblies with 16 burnable poison shims per assembly, 8 type E3 assemblies with 16 burnable poison shims per assembly, 4 type E4 assemblies with 8 burnable poison shims per assembly, and 20 type E5 assemblies with 12 burnable poison rods per assembly. These sub-batch types are fuel zone-enriched and their configurations are shown in Figure 3-2.

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

Figure 3-4 displays the beginning of Cycle 3 assembly average burnup distribution. The burnup distribution is based on a Cycle 2 length of 420 EFPD.

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 2

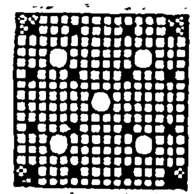
Cycle 3 Core Loading

Assembly Designation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Shims/ Assembly	Initial Shim Loading (gm B10/in)	Total Number of Fuel Rods	Total Number of Shim Rods
B	1	208 12	2.78 1.92	16	.018	208 12	16
C	36	224 12	3.30 2.78	0	---	8064 432	0
D0	32	184 52	4.02 3.57	0	---	5888 1664	0
D1	20	168 52	4.02 3.57	16	.022	3360 1040	320
D2	8	168 52	4.02 3.57	16	.020	1344 416	128
D3	16	168 52	3.57 3.09	16	.022	2688 832	256
D4	4	172 52	3.57 3.09	12	.008	688 208	48
D5	28	172 52	3.57 3.09	12	.020	4816 1456	336
E0	24	184 52	4.03 3.70	0	---	4416 1248	0
E1	16	168 52	4.03 3.70	16	.016	2688 832	256
E2	24	168 52	3.70 3.40	16	.020	4032 1248	384
E3	8	168 52	3.70 3.40	16	.016	1344 416	128
E4	4	176 52	4.03 3.70	8	.012	704 208	32
[illegible]	20	172 52	3.70 3.40	12	.020	3440 1040	240
TOTAL	241					54732	2144

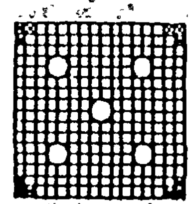
Figure 3-1
Loading of Assemblies Shuffled from Previous Cycle

Assembly Designation	Number of Assemblies	Fuel Rods per Assembly	Initial Enrichment (w/o U-235)	Number Shims/ Assembly	Initial Shim Loading (gm B10/in)
B	1	208 12	2.78 1.92	16	.018
C	36	224 12	3.30 2.78	0	---
D0	32	184 52	4.02 3.57	0	---
D1	20	168 52	4.02 3.57	16	.022
D2	8	168 52	4.02 3.57	16	.020
D3	16	168 52	3.57 3.09	16	.022
D4	4	172 52	3.57 3.09	12	.008
D5	28	172 52	3.57 3.09	12	.020

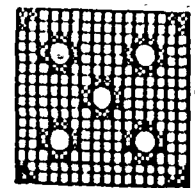
B



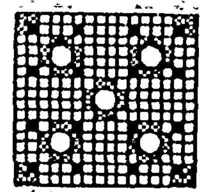
C



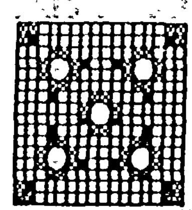
D0



D1
D2
D3



D4
D5



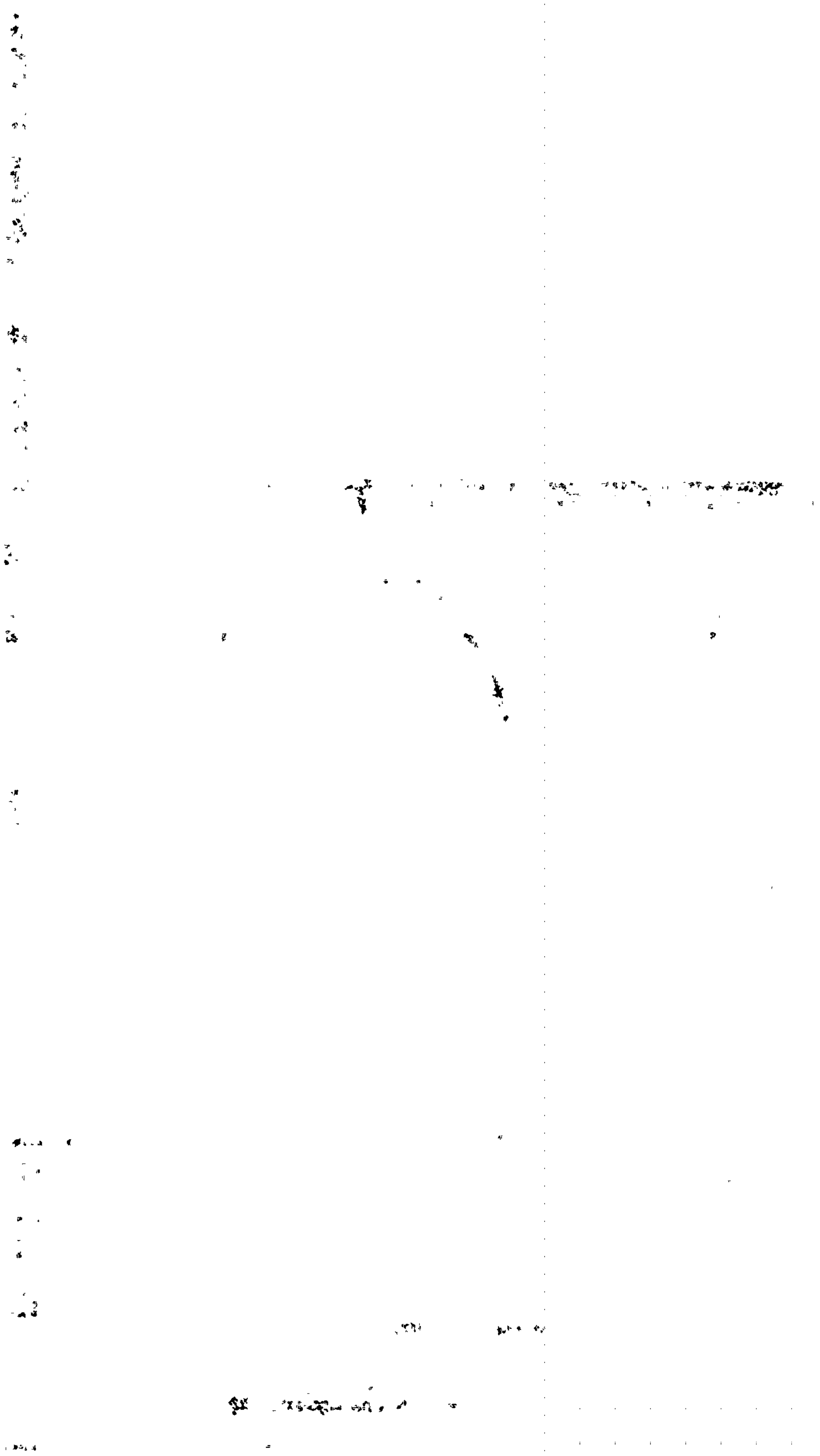
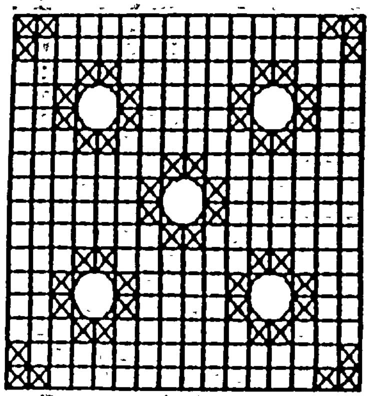


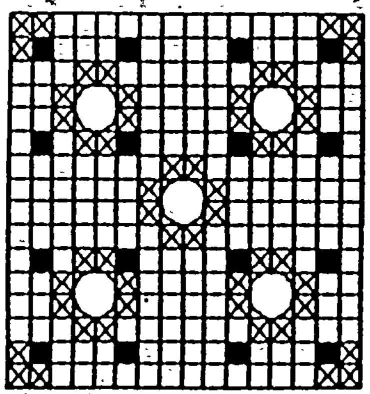
FIGURE 3-2
FRESH FEED ASSEMBLY FUEL LOADINGS
WATERHOLE AND SHIM PLACEMENT



SUB-BATCH E0 – 24 Assemblies

☐ 4.03 w/o U-235

☒ 3.70 w/o U-235

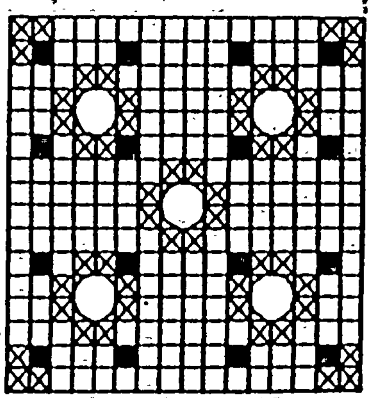


SUB-BATCH E1 – 16 Assemblies

☐ 4.03 w/o U-235

☒ 3.70 w/0 U-235

■ B_4C-AL_2O_3 Shim Pin 0.016 gm B-10/in

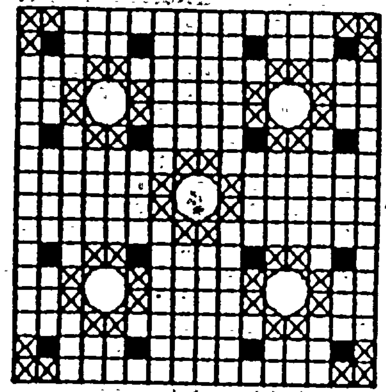


SUB-BATCH E2 – 24 Assemblies

☐ 3.70 w/o U-235

☒ 3.40 w/0 U-235

■ B_4C-AL_2O_3 Shim Pin 0.020 gm B-10/in

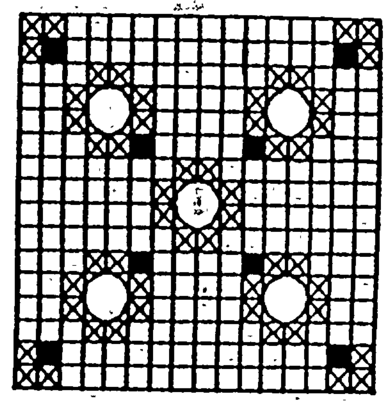


SUB-BATCH E3 – 8 Assemblies

☐ 3.70 w/o U-235

☒ 3.40 w/0 U-235

■ B_4C-AL_2O_3 Shim Pin 0.016 gm B-10/in

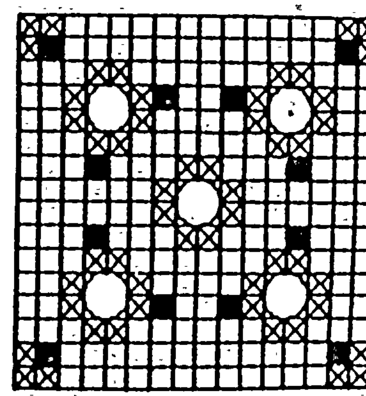


SUB-BATCH E4 – 4 Assemblies

☐ 4.03 w/o U-235

☒ 3.70 w/0 U-235

■ B_4C-AL_2O_3 Shim Pin 0.012 gm B-10/in



SUB-BATCH E5 – 20 Assemblies

☐ 3.70 w/o U-235

☒ 3.40 w/0 U-235

■ B_4C-AL_2O_3 Shim Pin 0.020 gm B-10/in

Figure 3-3
PVNGS Unit 2 Cycle 3 Fuel Management

					C	E0	D0	D1
			C	E0	D0	D0	D3	E4
		D2	E0	D1	E1	D5	E3	D5
	C	E0	C	E1	C	E2	D5	E5
	E0	D1	E1	D1	E5	D3	E2	D5
C	D0	E1	C	E5	D4	E5	C	D0
E0	D0	D5	E2	D3	E5	D5	E2	D2
D0	D3	E3	D5	E2	C	E2	D1	D0
D1	E4	D5	E5	D5	D0	D2	D0	B

Assy Type	# Shims	# Fuel Pins	Pin Enrichments & Zoning				Shim Loading, gm B-10/in	No. of Assy.	Avg. Assy. Enrichment
			# Pins	W/O	# Pins	W/O			
E0	0	236	184	4.03	52	3.70	--	24	3.957
E1	16	220	168	4.03	52	3.70	.016	16	3.952
E2	16	220	168	3.70	52	3.40	.020	24	3.629
E3	16	220	168	3.70	52	3.40	.016	8	3.629
E4	8	228	176	4.03	52	3.70	.012	4	3.955
E5	12	224	172	3.70	52	3.40	.020	20	3.630
								96	

Figure 3-4

PALO VERDE U2C3

					1 30,387	2 0	3 10,508	4 21,902
			5 29,853	6 0	7 16,056	8 13,589	9 22,227	10 0
		11 21,672	12 0	13 22,365	14 0	15 22,289	16 0	17 20,885
	18 29,840	19 0	20 30,589	21 0	22 24,069	23 0	24 22,271	25 0
	26 0	27 22,293	28 0	29 23,199	30 0	31 22,782	32 0	33 21,565
34 30,374	35 16,058	36 0	37 24,060	38 0	39 21,216	40 0	41 24,795	42 11,577
43 0	44 13,583	45 22,277	46 0	47 22,703	48 0	49 20,881	50 0	51 21,792
52 10,505	53 22,213	54 0	55 22,244	56 0	57 24,788	58 0	59 21,916	60 18,144
61 21,949	62 0	63 20,886	64 0	65 21,559	66 11,578	67 21,497	68 18,147	69 19,925

Assembly Average Burnup at BOC3
(MWD/T)

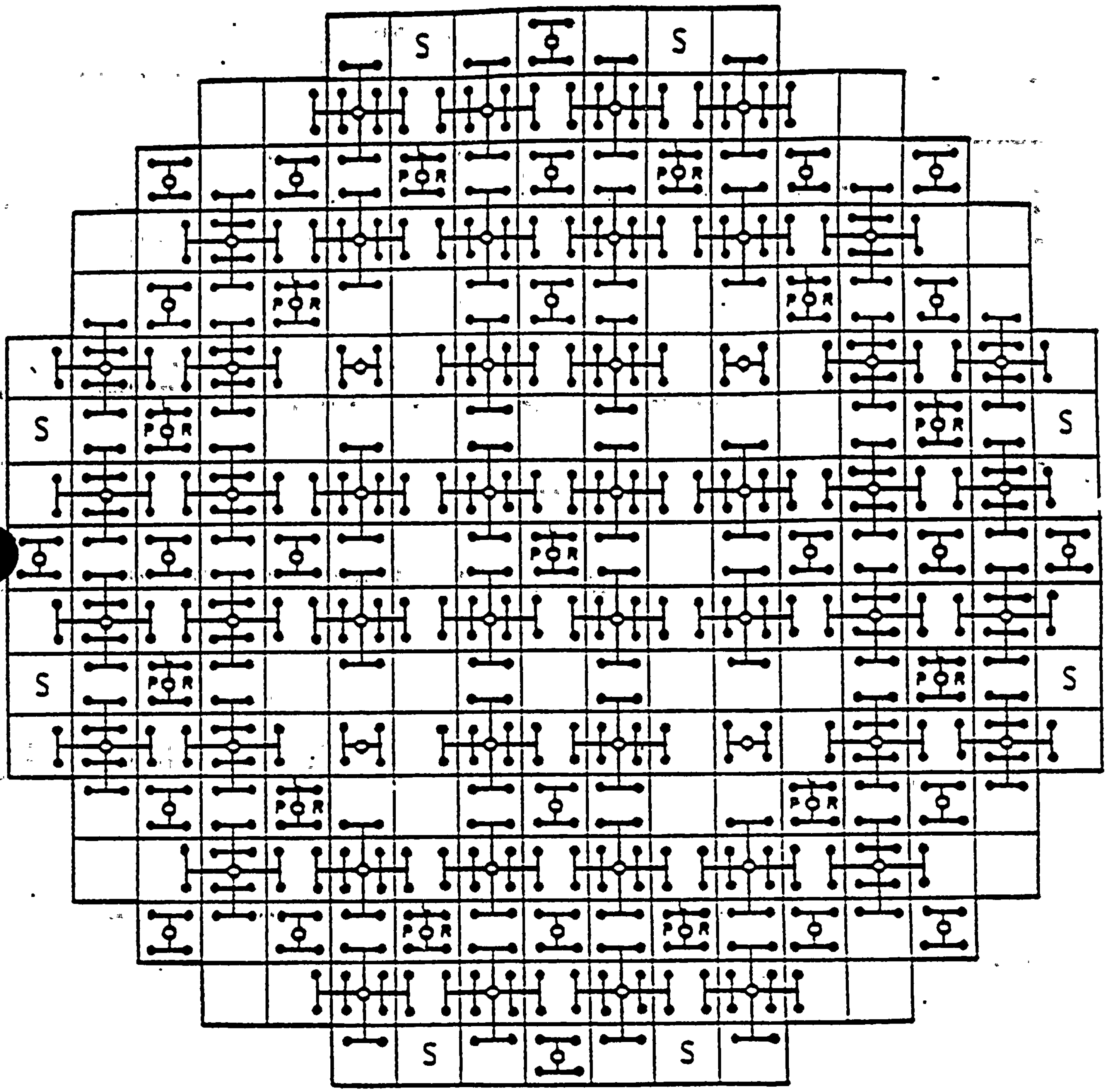
[illegible]

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

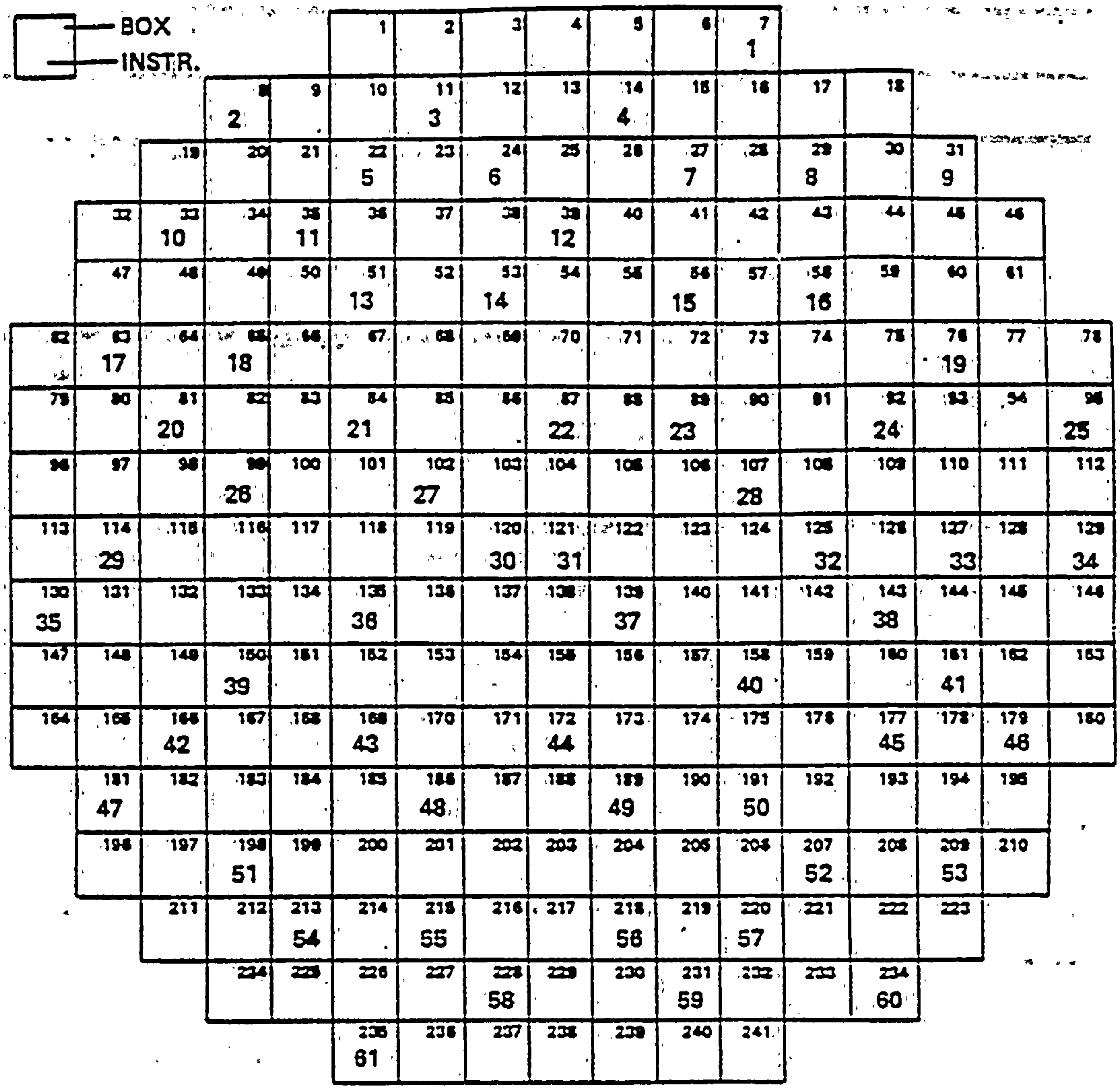
					1	2 S	3	4 3	5	6 S	7					
			8	9	10 A	11	12 1	13	14 1	15	16 A	17	18			
		19 4	20	21 2	22	23 P_2	24	25 3	26	27 P_2	28	29 2	30	31 4		
	32	33	34 B	35	36 B	37	38 B	39	40 B	41	42 B	43	44 B	45	46	
	47	48 2	49	50 P_1	51	52	53	54 5	55	56	57	58 P_1	59	60 2	61	
62	63 A	64	65 B	66	67 4	68	69 A	70	71 A	72	73 4	74	75 B	76	77 A	78
79 S	80	81 P_2	82	83	84	85	86	87	88	89	90	91	92	93 P_2	94	95 S
96	97 1	98	99 B	100	101 A	102	103 3	104	105 3	106	107 A	108	109 B	110	111 1	112
113 3	114	115 3	116	117 5	118	119	120	121 P_1	122	123	124	125 5	126	127 3	128	129 3
130	131 1	132	133 B	134	135 A	136	137 3	138	139 3	140	141 A	142	143 B	144	145 1	146
147 S	148	149 P_2	150	151	152	153	154	155	156	157	158	159	160	161 P_2	162	163 S
164	165 A	166	167 B	168	169 4	170	171 A	172	173 A	174	175 4	176	177 B	178	179 A	180
	181	182 2	183	184 P_1	185	186	187	188 5	189	190	191	192 P_1	193	194 2	195	
	196	197	198 B	199	200 B	201	202 B	203	204 B	205	206 B	207	208 B	209	210	
		211 4	212	213 2	214	215 P_2	216	217 3	218	219 P_2	220	221 2	222	223 4		
			224	225	226 A	227	228 1	229	230 1	231	232 A	233	234			
					235	236 S	237	238 3	239	240 S	241					

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

[illegible]

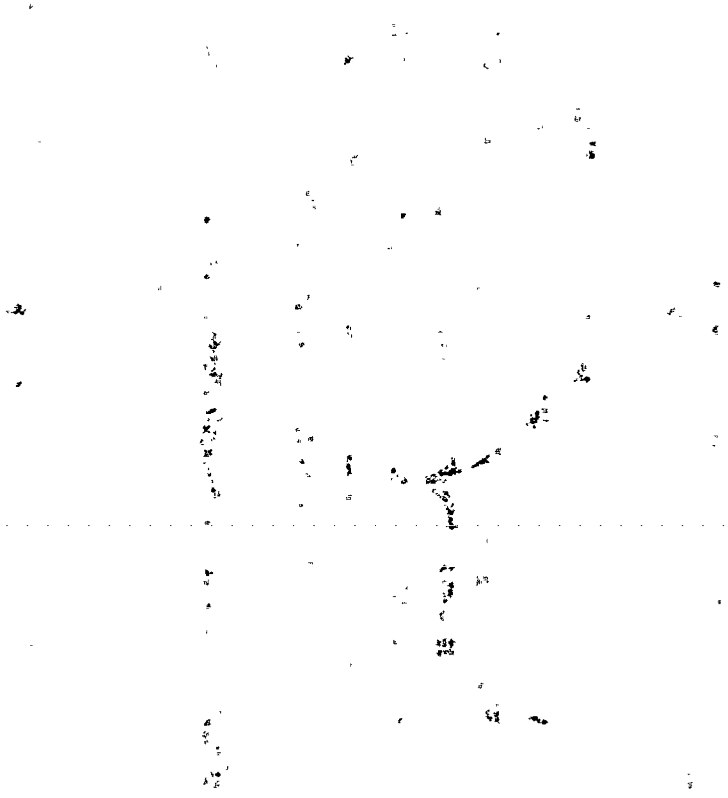


ARIZONA. Palo Verde Nuclear Generating Station	CEA PATTERN	Figure 3-5B



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

[illegible]



4.0 FUEL SYSTEM DESIGN

4.1 MECHANICAL DESIGN

4.1.1 Fuel Design

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

A design feature was incorporated into Batch E to improve the burnup capability of the poison rods. The poison rod assembly design was modified by increasing the overall length from 160.918 inches to 161.168 inches. This provides greater internal void volume which enables higher burnups with poison rods with higher B-10 loadings while reducing end of life internal pressure. In addition, this change makes the fuel and poison rods equal in length.

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.

4.2 GUIDE TUBE WEAR

Twenty of the fuel assemblies that had CEA's 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 and will not adversely affect the fuel assembly performance throughout its expected life in the core. Thus no guide tube wear inspections are necessary.

4.3 THERMAL DESIGN

The thermal performance of composite fuel pins that envelope the pins of fuel batches B, C, D and E present in Cycle 3 have been evaluated using the FATES3A version of the C-E fuel evaluation model (References 4-3 and 4-4) as approved by the NRC (Reference 4-5). FATES3A is the version of FATES3 that incorporates the grain size restriction given in Reference 4-5. 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 3.

4.4 CHEMICAL DESIGN

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for the Batch E fuel are essentially identical to those of the fuel batches included in Cycle 2. The exception being the reduction of the tin content in the fuel cladding for improved corrosion resistance. Thus, the chemical or metallurgical performance of the Batch E fuel is enhanced from the performance of the Cycle 2 fuel and thereby, providing the potential for higher burnup. Although the tin content has been reduced, it remains within the standard specification.

[illegible]

4.5 SHOULDER GAP ADEQUACY

Measured shoulder gap data acquired from post Cycle 1 inspection of fuel assemblies at PVNGS Units 1 and 2 confirmed the conservatism of the shoulder gap evaluation technique for the PVNGS fuel (references 4-1 and 4-2). This evaluation technique predicts adequate shoulder gaps for all fuel operating in Cycle 3.

5.0 NUCLEAR DESIGN

5.1 PHYSICS CHARACTERISTICS

5.1.1 Fuel Management

The Cycle 3 core makes use of a low-leakage fuel management scheme, in which previously burned Batch C assemblies are placed on the core periphery. Most of the fresh Batch E 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 2 endpoint at 420 EFPD, the Cycle 3 reactivity lifetime for full power operation is expected to be 430 EFPD. Explicit evaluations have been performed to assure applicability of all analyses to a Cycle 2 termination burnup of between 394 and 446 EFPD and for a Cycle 3 length up to 456 EFPD.

Characteristic physics parameters for Cycle 3 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 3 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.

[illegible]

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 CEA groups calculated at full power conditions for Cycle 3 and the Reference Cycle.

5.1.2 Power Distribution

Figures 5-3 through 5-5 illustrate the calculated All Rods Out (ARO) relative assembly power densities during Cycle 3. The one-pin planar radial power peaks (Fxy) presented in these figures represent the maximum over the mid eighty percent of the core axially. 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 configurations 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 1.21 at BOC3 to 1.11 at EOC3.

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. ROCS-DIT with the MC module will be used. ROCS-DIT and the MC module 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 3 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 2.

5.2.3 Nuclear Design Methodology

The Cycle 3 nuclear design was performed with two and three dimensional core models using the ROCS and MC computer codes employing DIT calculated cross sections. ROCS, MC, and DIT were described in Reference 5-2.

[illegible]

TABLE 5-1

PVNGS-2 CYCLE 3

NOMINAL PHYSICS CHARACTERISTICS

Dissolved Boron	Units	Reference Cycle	Cycle 3
Dissolved Boron Concentration for Criticality, CEAs Withdrawn, Hot Full Power Equilibrium Xenon, BOC	PPM	1116	1092
Boron Worth			
Hot Full Power, BOC	PPM/%$\Delta\rho$	120	121
Hot Full Power, EOC	PPM/%$\Delta\rho$	95	91
Moderator Temperature Coefficients			
Hot Full Power, Equilibrium Xenon			
Beginning of Cycle	$10^{-4}\Delta\rho/^{o}F$	-0.5	-0.6
End of Cycle	$10^{-4}\Delta\rho/^{o}F$	-2.0	-3.0
Hot Zero Power, Beginning of Cycle	$10^{-4}\Delta\rho/^{\circ}F$	---	+0.2
Doppler Coefficient			
Hot Zero Power, BOC	$10^{-5}\Delta\rho/^{o}F$	-1.8	-1.9
Hot Full Power, BOC	$10^{-5}\Delta\rho/^{o}F$	-1.4	-1.5
Hot Full Power, EOC	$10^{-5}\Delta\rho/^{o}F$	-1.7	-1.7
Total Delayed Neutron Fraction, β_{eff}			
BOC	-------	.0063	.0063
EOC	-------	.0052	.0051
Prompt Neutron Generation Time, l^*			
BOC	10^{-6} sec	21.7	20.7
EOC	10^{-6} sec	27.7	28.1

TABLE 5-2

PVNGS-2 CYCLE 3 LIMITING VALUES OF
REACTIVITY WORTHS AND ALLOWANCES FOR HOT
FULL POWER STEAM LINE BREAK, %$\Delta\rho$, END-OF-CYCLE (EOC)

		Reference Cycle	Cycle 3
1.	Worth of all CEAs Inserted	-16.0	-17.6
2.	Stuck CEA Allowance	+4.3	+4.0
3.	Worth of all CEAs Less Highest Worth CEA Stuck Out	-11.7	-13.6
4.	Full Power Dependent Insertion Limit CEA Bite	+0.2	+0.2
5.	Calculated Scram Worth	-11.5	-13.4
6.	Physics Uncertainty	+1.2	+1.3
7.	Other Allowances (losses due to voiding)	+0.1	+0.1
8.	Net Available Scram Worth	-10.2	-12.0
9.	Scram Worth Used in Safety Analysis	-10.0	-10.2

TABLE 5-3

PVNGS-2 CYCLE 3

REACTIVITY WORTH OF CEA REGULATING GROUPS

AT HOT FULL POWER, %Δρ

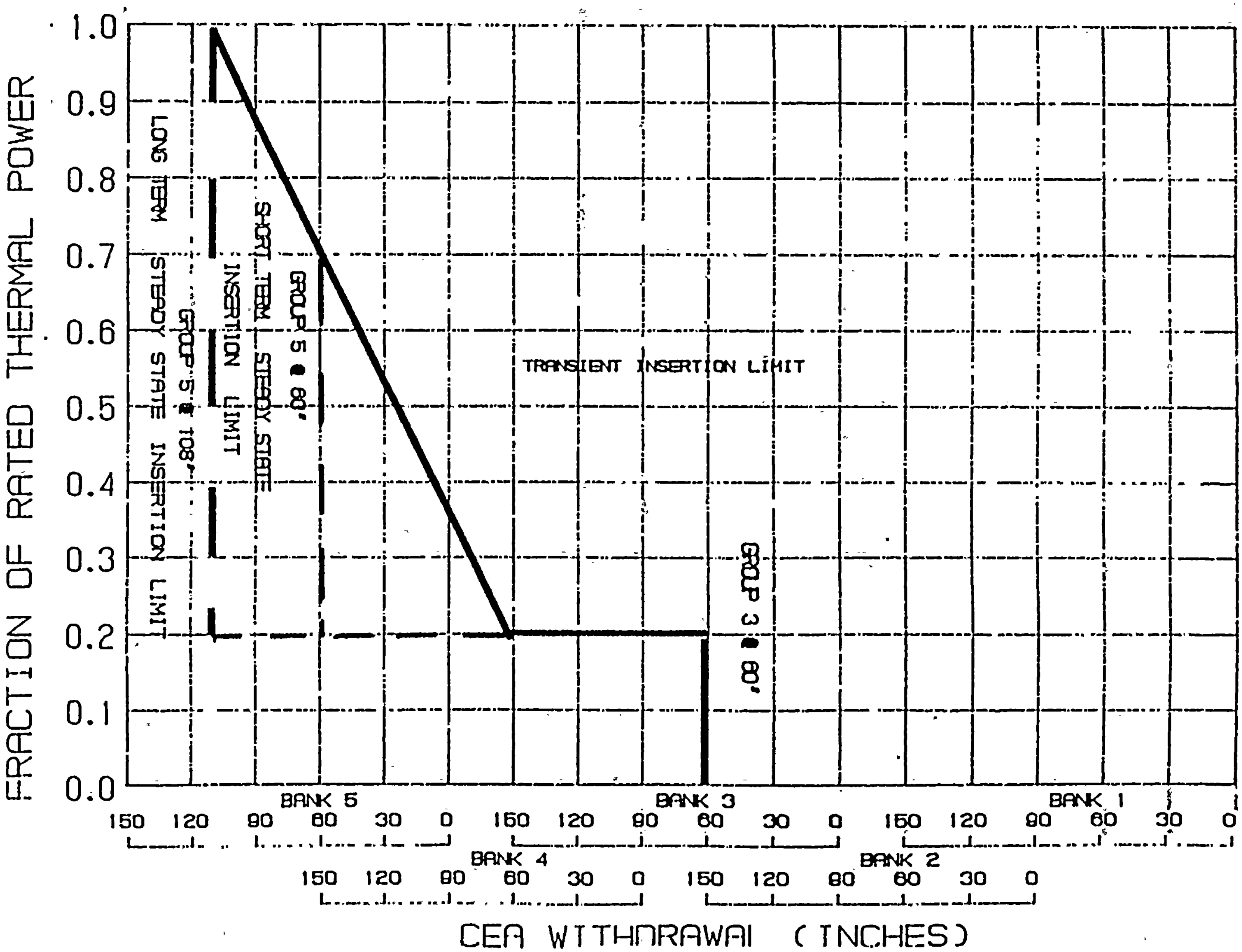
Regulating CEAs	Beginning of Cycle: Reference Cycle	Beginning of Cycle: Cycle 3	End of Cycle: Reference Cycle	End of Cycle: Cycle 3
Group 5	-.25	-.28	-.29	-.29
Group 4	-.39	-.41	-.46	-.41
Group 3	-.67	-.91	-.74	-.88

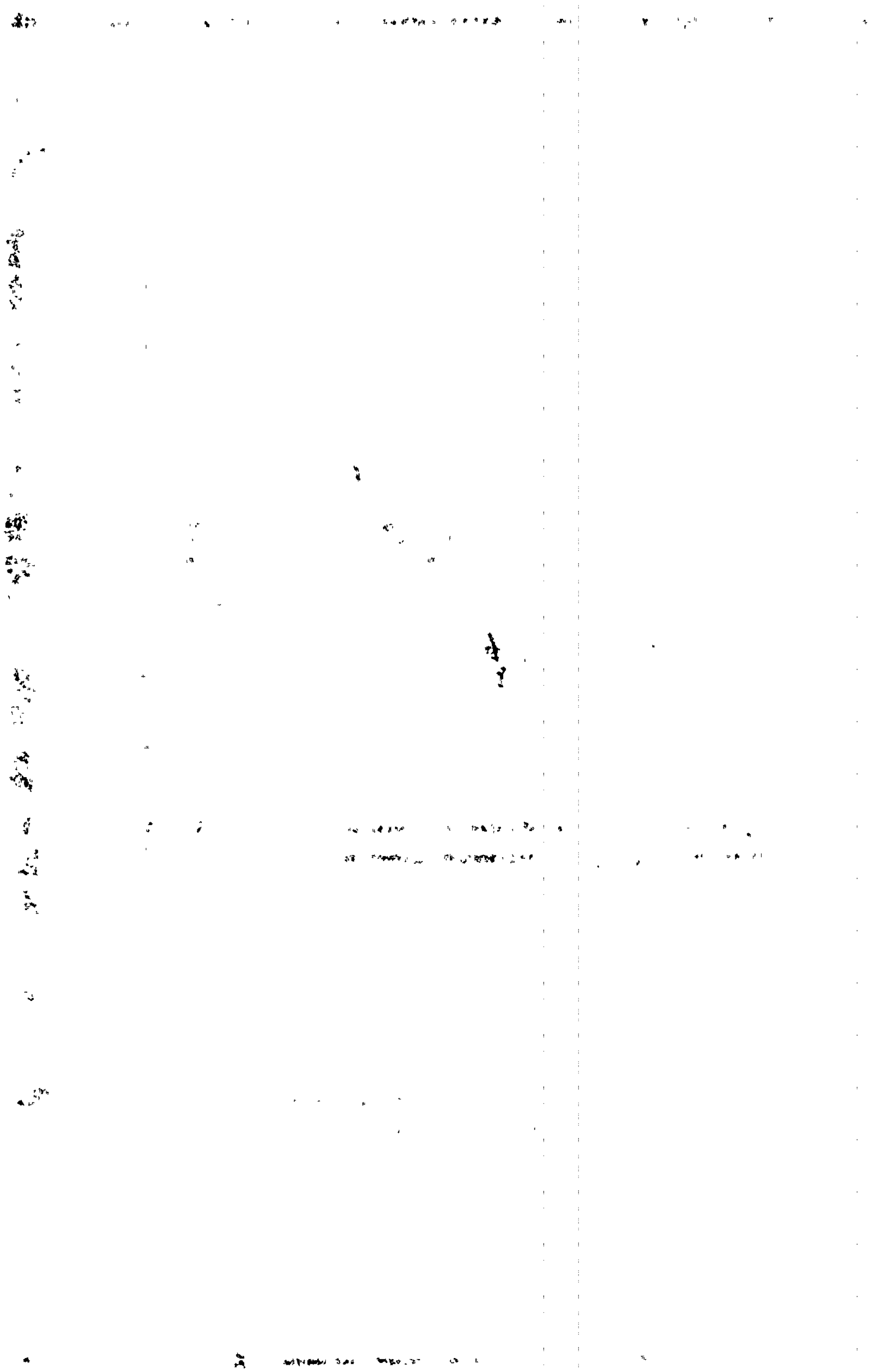
Note:

Values shown assume sequential group insertion.

FIGURE 5-1

CEA INSERTION LIMITS VS. THERMAL POWER





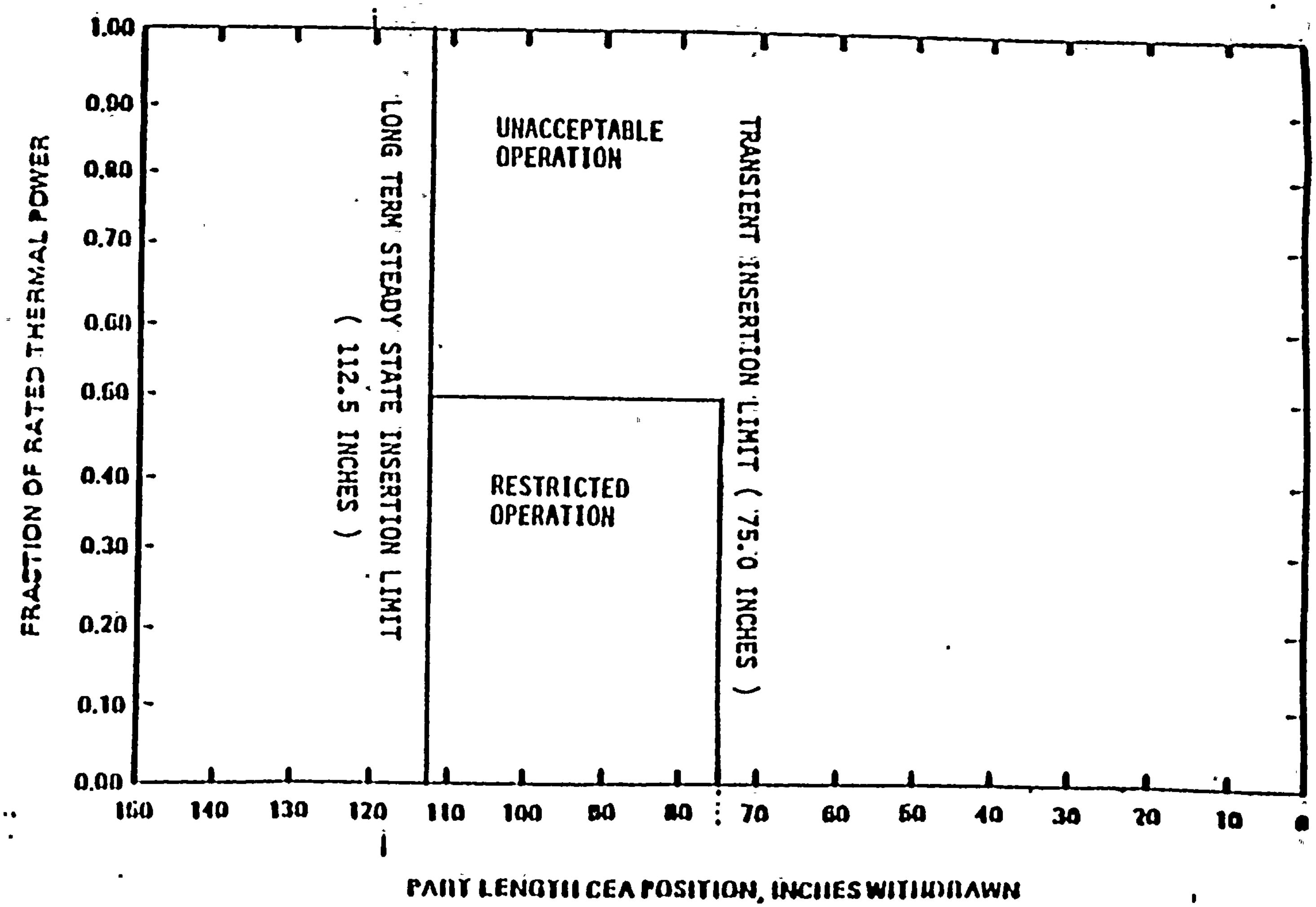


FIGURE 5-2

PART LENGTH CEA INSERTION LIMIT VS THERMAL POWER

Figure 5-3

PALO VERDE U2C3

F_{xy}=1.532
Box 50

					1 0.3027	2 0.6978	3 0.6328	4 0.5771
			5 0.3515	6 0.8511	7 0.8627	8 0.9748	9 0.9041	10 1.1504
		11 0.5154	12 1.0368	13 1.0256	14 1.1985	15 0.9851	16 1.1657	17 1.0342
	18 0.3521	19 1.0379	20 0.8198	21 1.2307	22 0.9396	23 1.1768	24 1.0308	25 1.2433
	26 0.8525	27 1.0277	28 1.2319	29 1.1450	30 1.2928	31 1.0827	32 1.2181	33 1.0808
34 0.3034	35 0.8644	36 1.2007	37 0.9411	38 1.2939	39 1.1465	40 1.3249	41 1.0055	42 1.2895
43 0.6996	44 0.9771	45 0.9873	46 1.1790	47 1.0849	48 1.3256	49 1.1597	50 1.3437	51 1.2813
52 0.6349	53 0.9068	54 1.1685	55 1.0332	56 1.2203	57 1.0068	58 1.3442	59 1.2654	60 1.2468
61 0.5812	62 1.1540	63 1.0368	64 1.2459	65 1.0828	66 1.2915	67 1.2854	68 1.2465	69 1.0703

ARO Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
BOC3

Figure 5-4

PALO VERDE U2C3

F_{xy}=1.488
Box 30

					1 0.3045	2 0.6674	3 0.6261	4 0.5917
			5 0.3386	6 0.7955	7 0.8351	8 0.9413	9 0.9068	10 1.1646
		11 0.4871	12 0.9461	13 0.9987	14 1.2672	15 1.0148	16 1.2779	17 1.0685
	18 0.3390	19 0.9467	20 0.8071	21 1.3053	22 0.9814	23 1.3034	24 1.0819	25 1.3194
	26 0.7960	27 1.0001	28 1.3060	29 1.1626	30 1.3412	31 1.1072	32 1.3071	33 1.0776
34 0.3046	35 0.8356	36 1.2682	37 0.9822	38 1.3418	39 1.1257	40 1.3355	41 0.9758	42 1.1765
43 0.6676	44 0.9418	45 1.0155	46 1.3045	47 1.1088	48 1.3358	49 1.1132	50 1.3173	51 1.1445
52 0.6265	53 0.9074	54 1.2786	55 1.0829	56 1.3081	57 0.9763	58 1.3172	59 1.1151	60 1.0483
61 0.5936	62 1.1655	63 1.0690	64 1.3202	65 1.0783	66 1.1774	67 1.1478	68 1.0477	69 0.8866

ARO Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
MOC3

Figure 5-5

PALO VERDE U2C3

F_{xy}=1.467
Box 28

					1 0.3383	2 0.6990	3 0.6616	4 0.6317
			5 0.3684	6 0.8127	7 0.8561	8 0.9494	9 0.9192	10 1.1646
		11 0.5203	12 0.9530	13 1.0055	14 1.2998	15 1.0254	16 1.2943	17 1.0597
	18 0.3688	19 0.9536	20 0.8325	21 1.3238	22 0.9964	23 1.3397	24 1.0800	25 1.3174
	26 0.8128	27 1.0063	28 1.3239	29 1.1370	30 1.3209	31 1.0888	32 1.3184	33 1.0536
34 0.3381	35 0.8557	36 1.2993	37 0.9963	38 1.3208	39 1.0814	40 1.2984	41 0.9534	42 1.1034
43 0.6983	44 0.9487	45 1.0246	46 1.3388	47 1.0891	48 1.2981	49 1.0647	50 1.2750	51 1.0651
52 0.6609	53 0.9183	54 1.2927	55 1.0785	56 1.3174	57 0.9531	58 1.2745	59 1.0368	60 0.9651
61 0.6322	62 1.1633	63 1.0581	64 1.3150	65 1.0526	66 1.1032	67 1.0675	68 0.9645	69 0.8254

ARO Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
EOC3

Figure 5-6

PALO VERDE U2C3

F_{xy}=1.512
Box 30

					1 0.3162	2 0.7384	3 0.6659	4 0.6034
			5 0.3694	6 0.9130	7 0.9215	8 1.0370	9 0.9461	10 1.1982
		11 0.5522	12 1.1241	13 1.1003	14 1.2630	15 1.0209	16 1.1679	17 1.0289
	18 0.3701	19 1.1253	20 0.8678	21 1.2984	22 0.9601	23 1.1671	24 0.9637	25 1.0914
	26 0.9147	27 1.1028	28 1.2997	29 1.1976	30 1.3195	31 1.0562	32 1.0539	33 0.6350
34 0.3170	35 0.9236	36 1.2657	37 0.9618	38 1.3207	39 1.1598	40 1.2937	41 0.9153	42 1.1403
43 0.7405	44 1.0399	45 1.0236	46 1.1697	47 1.0585	48 1.2945	49 1.1371	50 1.3010	51 1.2443
52 0.6683	53 0.9494	54 1.1713	55 0.9664	56 1.0559	57 0.9164	58 1.3015	59 1.2622	60 1.2549
61 0.6078	62 1.2027	63 1.0322	64 1.0942	65 0.6363	66 1.1420	67 1.2480	68 1.2546	69 1.0421

Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
BOC3 with Bank 5 Inserted

Figure 5-7

PALO VERDE U2C3

F_{xy}=1.630
B0X 50

					1 0.2911	2 0.6875	3 0.6309	4 0.5779
			5 0.3370	6 0.8345	7 0.8434	8 0.9553	9 0.9042	10 1.1660
		11 0.5029	12 1.0244	13 1.0059	14 1.1548	15 0.8868	16 1.1515	17 1.0508
	18 0.3376	19 1.0256	20 0.7880	21 1.1784	22 0.9008	23 1.1496	24 1.0432	25 1.2731
	26 0.8362	27 1.0082	28 1.1797	29 1.0226	30 1.2703	31 1.0981	32 1.2519	33 1.1308
34 0.2919	35 0.8453	36 1.1573	37 0.9024	38 1.2715	39 1.1695	40 1.3739	41 1.0459	42 1.3859
43 0.6894	44 0.9581	45 0.8891	46 1.1521	47 1.1006	48 1.3748	49 1.2246	50 1.4255	51 1.3772
52 0.6333	53 0.9074	54 1.1548	55 1.0461	56 1.2543	57 1.0473	58 1.4260	59 1.3522	60 1.3258
61 0.5821	62 1.1703	63 1.0540	64 1.2762	65 1.1331	66 1.3882	67 1.3814	68 1.3256	69 1.0047

Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
BOC3 with PLCEAs Inserted

Figure 5-8

PALO VERDE U2C3

F_{xy}=1.566
Box 50

					1 0.3146	2 0.7358	3 0.6679	4 0.6079
			5 0.3720	6 0.9134	7 0.9097	8 1.0124	9 0.9405	10 1.2031
		11 0.5569	12 1.1285	13 1.0952	14 1.2331	15 0.9179	16 1.1505	17 1.0327
	18 0.3727	19 1.1297	20 0.8610	21 1.2673	22 0.9427	23 1.1489	24 0.9677	25 1.1064
	26 0.9152	27 1.0977	28 1.2687	29 1.0776	30 1.3010	31 1.0702	32 1.0820	33 0.6553
34 0.3155	35 0.9118	36 1.2357	37 0.9443	38 1.3022	39 1.1727	40 1.3342	41 0.9544	42 1.1936
43 0.7379	44 1.0153	45 0.9204	46 1.1514	47 1.0726	48 1.3350	49 1.1860	50 1.3629	51 1.3037
52 0.6704	53 0.9438	54 1.1539	55 0.9704	56 1.0841	57 0.9556	58 1.3634	59 1.3140	60 1.2919
61 0.6123	62 1.2077	63 1.0360	64 1.1092	65 0.6566	66 1.1955	67 1.3077	68 1.2916	69 0.9851

Assembly Relative Power Densities
at Hot Full Power with Eq. Xe
BOC3 with Bank 5 and PLCEAs Inserted

Figure 5-9

PALO VERDE U2C3

F_{xy}=1.555
Box 21

					1 0.3629	2 0.7766	3 0.7121	4 0.6650
			5 0.4044	6 0.9194	7 0.9300	8 1.0126	9 0.9508	10 1.2308
		11 0.5783	12 1.0826	13 1.0903	14 1.4182	15 1.0506	16 1.3126	17 1.0324
	18 0.4046	19 1.0828	20 0.9023	21 1.4443	22 1.0287	23 1.3538	24 0.9874	25 1.1550
	26 0.9192	27 1.0910	28 1.4443	29 1.1717	30 1.3504	31 1.0294	32 1.1393	33 0.5890
34 0.3628	35 0.9295	36 1.4176	37 1.0285	38 1.3503	39 1.0507	40 1.2451	41 0.8373	42 0.9131
43 0.7759	44 1.0119	45 1.0500	46 1.3532	47 1.0299	48 1.2449	49 0.9915	50 1.1971	51 0.9578
52 0.7116	53 0.9503	54 1.3115	55 0.9870	56 1.1389	57 0.8372	58 1.1966	59 0.9564	60 0.8927
61 0.6660	62 1.2300	63 1.0314	64 1.1541	65 0.5888	66 0.9130	67 0.9598	68 0.8921	69 0.7637

Assembly Relative Power Densities at Full Power
EOC3 with Bank 5 Inserted

[illegible]

Figure 5-10

PALO VERDE U2C3

F_{xy}=1.466
Box 32

					1 0.3343	2 0.7244	3 0.6781	4 0.6410
			5 0.3697	6 0.8405	7 0.8480	8 0.9269	9 0.9104	10 1.2033
		11 0.5277	12 0.9869	13 0.9946	14 1.2867	15 0.8921	16 1.2898	17 1.0574
	18 0.3698	19 0.9871	20 0.8157	21 1.2991	22 0.9591	23 1.3265	24 1.0686	25 1.3538
	26 0.8403	27 0.9952	28 1.2991	29 0.9741	30 1.2925	31 1.0774	32 1.3691	33 1.0724
34 0.3342	35 0.8475	36 1.2861	37 0.9590	38 1.2924	39 1.0658	40 1.3458	41 0.9799	42 1.1478
43 0.7238	44 0.9263	45 0.8916	46 1.3258	47 1.0779	48 1.3455	49 1.0961	50 1.3538	51 1.0971
52 0.6776	53 0.9098	54 1.2887	55 1.0682	56 1.3684	57 0.9797	58 1.3533	59 1.0611	60 0.9738
61 0.6419	62 1.2024	63 1.0564	64 1.3528	65 1.0719	66 1.1476	67 1.0994	68 0.9731	69 0.7445

Assembly Relative Power Densities at Full Power
EOC3 with PLCEAs Inserted

[illegible]

Figure 5-11

PALO VERDE U2C3

F_{xy}=1.519
Box 21

					1 0.3616	2 0.7757	3 0.7178	4 0.6742
			5 0.4086	6 0.9204	7 0.9147	8 0.9814	9 0.9454	10 1.2408
		11 0.5854	12 1.0881	13 1.0826	14 1.3732	15 0.9219	16 1.2875	17 1.0385
	18 0.4087	19 1.0883	20 0.8917	21 1.3981	22 1.0037	23 1.3253	24 0.9927	25 1.1767
	26 0.9202	27 1.0833	28 1.3980	29 1.0278	30 1.3235	31 1.0471	32 1.1798	33 0.6140
34 0.3614	35 0.9142	36 1.3726	37 1.0035	38 1.3234	39 1.0660	40 1.2998	41 0.8875	42 0.9735
43 0.7751	44 0.9808	45 0.9214	46 1.3247	47 1.0476	48 1.2996	49 1.0538	50 1.2825	51 1.0273
52 0.7173	53 0.9449	54 1.2865	55 0.9923	56 1.1794	57 0.8874	58 1.2820	59 1.0192	60 0.9373
61 0.6753	62 1.2400	63 1.0376	64 1.1758	65 0.6137	66 0.9734	67 1.0294	68 0.9366	69 0.7210

Assembly Relative Power Densities at Full Power
EOC3 with Bank 5 and PLCEAs Inserted

6.0 THERMAL-HYDRAULIC DESIGN

6.1 DNBR ANALYSIS

Steady state DNBR analyses of Cycle 3 at the rated power level of 3800 MWT have been performed using the TORC computer 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-2 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 3 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 3 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 3 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 30,000 MWD/MTU. 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 3.

[illegible]

TABLE 6-1
PVNGS-2 Cycle 3
Thermal Hydraulic Parameters at Full Power

General Characteristics	Units	Reference Cycle	Cycle 3
Total Heat Output (Core only)	MWt 10^6 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)	^oF	565.0	565.0
Total Reactor Coolant Flow++ (Minimum Steady State)	gpm 10^6 lb/hr	423,300 155.8	423,300 155.8
Coolant Flow Through Core (Minimum)	10^6 lb/hr	151.1	151.1
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039
Average Mass Velocity	10^6 lb/hr-ft	2.49	2.49
Pressure Drop Across Core (Minimum steady state flow irreversible ΔP over entire fuel assembly)	psi	14.5	14.5
Total Pressure Drop Across Vessel (Based on nominal dimensions and minimum steady state flow)	psi	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	186,600*	185,100**
Total Heat Transfer Area (Accounts for axial densification factor)	ft^2	67,700*	68,300**
Film Coefficient at Average Conditions	BTU/hr-ft^2 ^oF	6100	6100
Average Film Temperature Difference	^oF	31	30
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod)	kw/ft	5.5	5.4

TABLE 6-1 (continued)

General Characteristics	Units	Reference Cycle	Cycle 3
Average Core Enthalpy Rise	BTU/lb	85.9	85.9
Maximum Clad Surface Temperature	^{o}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 2576 poison rods.

** Based on 2144 poison rods.

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

++ Tech. Spec. minimum flow rate.

7.0 NON-LOCA SAFETY ANALYSIS

7.0.1 Introduction

This section presents the results of the Palo Verde Nuclear Generating Station Unit 2 (PVNGS-2), Cycle 3 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 3. 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-2 Cycle 3 is the same as the Cycle 2 (Reference Cycle) methodology (References 7-1, 7-2 and 7-9) unless otherwise stated in the event presentations. Only methodology that has previously been reviewed and approved on the PVNGS dockets (References 7-10 and 7-11), the CESSAR docket (Reference 7-2), or on other dockets is used.

[illegible]

7.0.3 Mathematical Models

The mathematical models and computer codes used in the Cycle 3 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 described 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 only exceptions are the Shaft Seizure and Sheared Shaft 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 and 7-9.

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.

[illegible]

[illegible]

7.0.4 Input Parameters and Analysis Assumptions

Table 7.0-6 summarizes the core parameters assumed in the Cycle 3 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. Tech Spec changes 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 3 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 Cycle 3 Technical Specifications.

7.0.5 Conclusion

All DBEs have been evaluated for PVNGS-2, Cycle 3 to determine whether their results are bounded by the Reference Cycle.

Table 7.0-1

PVNGS Unit 2 Design Basis
Events Considered in the Cycle 3 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 System 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

[illegible]

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
7.4.6	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	Presented *
7.3.1	6) Total Loss of Forced Reactor Coolant Flow	Bounded by Reference Cycle
7.4.1	7) Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition	Bounded by Reference Cycle
7.4.2	8) Uncontrolled CEA Withdrawal at Power	Bounded by Reference Cycle
7.4.3	9) CEA Misoperation Events	Bounded by Reference Cycle
7.6.1	10) Pressurizer Pressure Decrease Events	Bounded by Reference Cycle
7.7.1	11) 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

* The Base Case is bounded by Reference Cycle. Results of the Event with Loss of Offsite Power is presented.

[illegible]

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 2, Cycle 3
Core Parameters Input to Safety Analyses

Safety Parameters	Units	Reference Cycle Value	Cycle 3 Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	3898	3898
Core Inlet Steady State Temperature	^oF	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 for All Powers	ASI Units	-0.3 to +0.3	-0.3 to +0.3
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

[illegible]

[illegible]

[illegible]

Table 7.0-6 (continued)

Safety Parameters	Units	Reference Cycle Value	Cycle 3 Value
Minimum DNBR CE-1 (SAFDL) Macbeth (Fuel failure limit for post-trip SLB with LOAC - References 7-5 and 7-6)		 1.24 1.30	 1.24 1.30
Initial Moderator Temperature Coefficient	10^{-4} $\Delta\rho/^{o}F$	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 amount of predicted failed fuel has increased for the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with Loss of Offsite Power after Turbine Trip (IOSGADV+LOP) from 8% to 12%. The increase in failed fuel was the result of more adverse nuclear power distributions. All other reference cycle data related to the IOSGADV remain applicable.

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

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 4.5%. As with section 7.1.4, 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.

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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 Event

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.

[illegible]

[illegible]

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 Pipe 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.

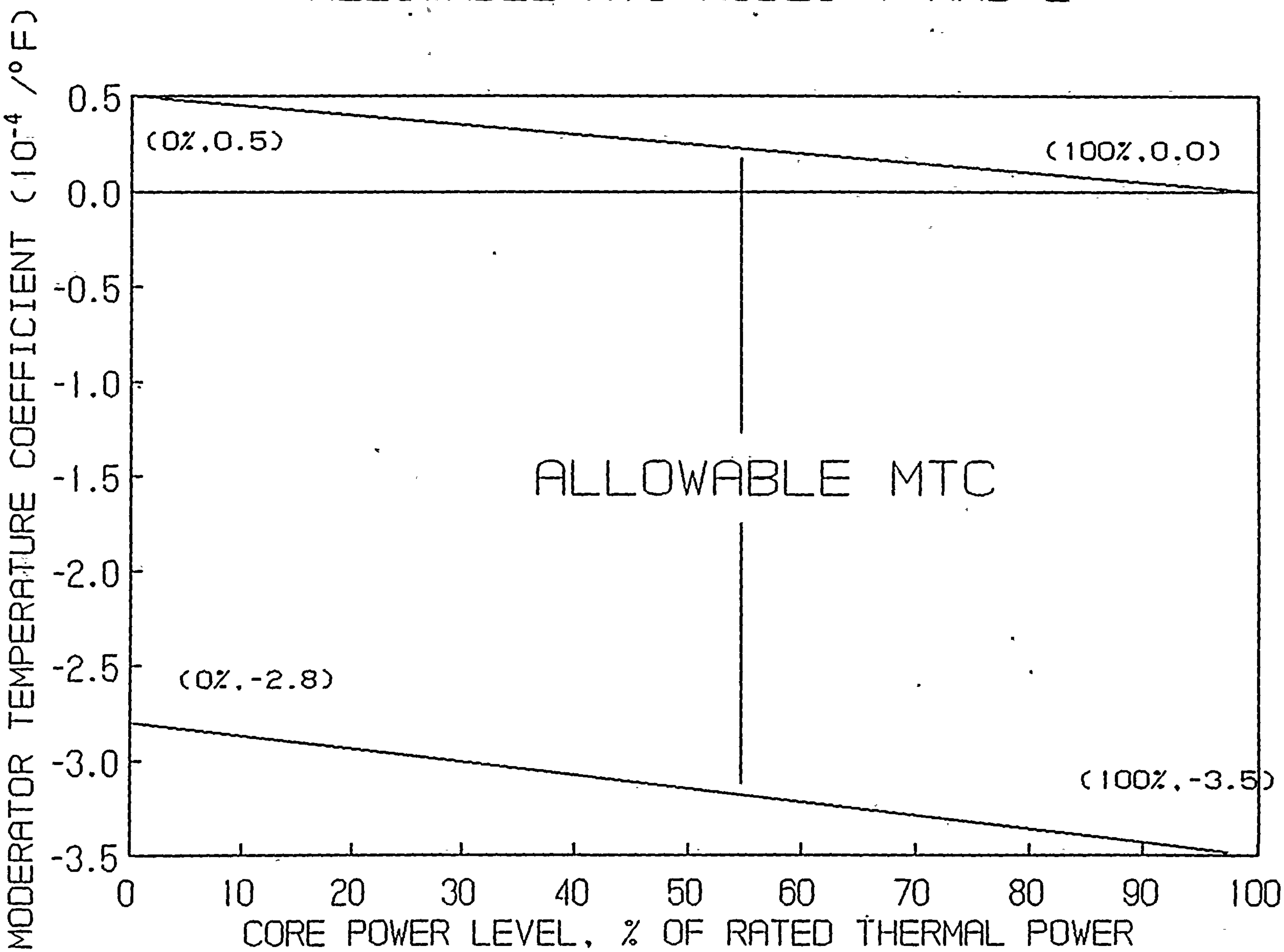
7.7 MISCELLANEOUS

7.7.1 Asymmetric Steam Generator Events

The results are bounded by the Reference Cycle.

FIGURE [illegible]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-2 Cycle 3 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 ECCS performance analysis for PVNGS-2 Cycle 3 consisted of an evaluation of the differences between Cycle 3 and PVNGS-2 Cycle 1. For this reason PVNGS-2, Cycle 1 shall be referred to as the Reference Cycle in Section 8. 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-2 Cycle 3 since there have been no significant adverse changes to RCS or ECCS hardware characteristics, or to core and system parameters. Therefore, only fuel rod clad temperature and oxidation calculations are required to re-evaluate ECCS performance with respect to the changes in fuel conditions introduced by Cycle 3. The NRC approved STRIKIN-II (Reference 8-5) code was used for this purpose.

[illegible]

Burnup dependent calculations were performed with STRIKIN-II to determine the limiting conditions for the ECCS performance analysis. The fuel performance data was generated with the FATES-3A fuel evaluation model (References 8-6 and 8-7) with the NRC grain size restriction (Reference 8-8). 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 Unit 2 Cycle 3 analysis considered a reduction of 470 gpm in LPSI runout flow relative to the Reference Cycle. The evaluation confirmed that there is adequate safety injection flow to maintain a full downcomer with the reduced flow. Therefore, this reduction in LPSI flow will not affect the results.

The acceptable performance of Unit 2 Cycle 3 has also been confirmed with up to 400 plugged tubes per steam generator and with a reduction in system flow rate to 155.8 X 10^6 lbm/hr and a reduction in core flow rate to 151.1 X 10^6 lbm/hr. For the Unit 2 Cycle 3 analysis nuclear flux augmentation factors were set to unity. The allowable PLHGR of 13.5 kw/ft for Cycle 3 is a reduction of 0.5 kw/ft from the Reference Cycle.

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 the hydraulics are identical, is the limiting break size for Cycle 3.

8.1.3 Results

The ECCS performance analysis for PVNGS-2 Cycle 3 showed that the reference analysis results conservatively apply. The peak clad temperature, maximum local clad 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.

8.1.4 Conclusion

Conformance to the ECCS criteria is demonstrated by the analysis results. Therefore, operation of PVNGS-2 Cycle 3 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

A review of Cycle 3 fuel and core data confirmed that the reported small break loss-of-coolant accident results (Reference 8-9) for PVNGS-2 Cycle 1 bounds PVNGS-2 Cycle 3. Therefore, acceptable small break LOCA 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). This acceptable performance has been confirmed with up to 400 plugged tubes per steam generator.

The reduction in delivered low pressure safety injection flow (see Reference 8-10) does not impact the small break loss-of-coolant analysis. The fuel cladding temperature excursion is either terminated by the high pressure safety injection pump flow or by the discharge of the safety injection tanks.

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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.

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10.0 TECHNICAL SPECIFICATIONS

This section provides a summary of the proposed changes to the Technical Specifications for PVNGS-2 Cycle 3. The changes are arranged in numerical order. Detailed change pages for the Technical Specifications are presented elsewhere.

Section	Title	Nature of Change
3.1.1.2, Figure 3.1-1A	Shutdown Margin T_{avg} greater than 210°F	Revise Shutdown Margin Requirements consistent with Cycle 3 analyses.
3.1.2.7, Tables 3.1-2, 3.1-3, and 3.1-5	Boron Dilution Alarms	Revise Tables 3.1-2, 3.1-3, and 3.1-5 on monitoring frequency with 1 or 2 boron dilution alarms inoperable, due to increased BOC critical boron concentrations for Cycle 3.
3.1.3.6 Figures 3.1-3, 3.1-4	Regulating CEA Insertion Limit	Revise Figures 3.1-3 and 3.1-4 to reflect Cycle 3 core characteristics.
3.2.4 Figures 3.2-2 and 3.2-2A	DNBR Margin	Revise Figures 3.2-3 and 3.2-2A to reflect Cycle 3 core characteristics.
3.2.7	Axial Shape Index	Change COLSS ASI limits to $\pm$0.27.

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 ANSI/ANS-19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors" 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

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. The boron concentration of the reactor coolant would then be reduced until criticality is attained. By the second method, the boron concentration is adjusted to the expected critical concentration with the shutdown and Part-Length CEA groups fully withdrawn. The regulating CEA groups would be withdrawn 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 ±1% Δk/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 and at a partially rodded configuration. The coolant temperature will be varied and the resulting reactivity change will be measured. The measured values will be verified to be within $\pm 0.3 \times 10^{-4}$ $\Delta k/k/°F$ 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 ±10% of their predicted values.

11.2.3 <u>Shape Annealing Matrix (SAM) and Boundary Point Power Correlation Coefficients (BPPCC) Verification</u>

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 <u>Radial Peaking Factor (RPF) and CEA Shadowing Factor (RSF) Verification</u>

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.

[illegible]

11.2.5 Temperature Reactivity Coefficients at Power

The isothermal temperature coefficient (ITC) will be measured at approximately full power. The ITC will be measured by changing coolant temperature, compensating with CEA motion, and maintaining power steady. The ITC will be verified to be within $\pm 0.3 \times 10^{-4}$ $\Delta k/k/°F$ of the predicted value.

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 ±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

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(1-2) 161-00683-EEVB/LJM, "Palo Verde Nuclear Generating Station Unit No. 2 Docket No. STN 50-529, Submittal of the Reload Analysis Report for Unit 2 Cycle 2," December 3, 1987.

12.2 SECTION 2.0 REFERENCES

None

12.3 SECTION 3.0 REFERENCES

None

12.4 SECTION 4.0 REFERENCES

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

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(4-5) R. A. Clark (NRC) to A. E. Lundvall, Jr. (BG&E), "Safety Evaluation of CEN-161 (FATES3)," March 31, 1983.

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(4-8) CESSAR SSER2, Section 4.2.5, "Guide Tube Wear Surveillance".

12.5 SECTION 5.0 REFERENCES

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(5-2) CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April, 1983.

12.6 SECTION 6.0 REFERENCES

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(6-2) CENPD-162-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September, 1976.

(6-3) 161-01867-DBK/JRP, "Generic Applicability of the CETOP-D Code for PWR Core T-H Analysis," April 26, 1989.

(6-4) CEN-356(V)-P-A, Rev. 01-P-A, "Modified Statistical Combination of Uncertainties", May, 1988.

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(6-6) CESSAR SSER 2 Section 4.4.6, Statistical Combination of Uncertainties.

(6-7) CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.

(6-8) CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-uniform Axial Power Distribution," December, 1984.

12.7 SECTION 7.0 REFERENCES

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(7-2) "CESSAR, Combustion Engineering Standard Safety Analysis Report," Docket No. 50-470.

(7-3) "CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981, Enclosure 1-P to LD-82-001, January 6, 1982.

(7-4) 161-01867-DBK/JRP, "Generic Applicability of the CETOP-D Code for PWR Core T-H Analysis," April 26, 1989.

(7-5) CENPD-188-A, "HERMITE Space-Time Kinetics," July, 1975.

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12.8 SECTION 8.0 REFERENCES-ECCS ANALYSIS

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161-01155-EEVB/BJA, dated July 6, 1988, "LPSI Flow Requirements".

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12.9 SECTION 9.0 REFERENCES

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(9-5) CEN-356(V)-P-A, Rev. 01-P-A, "Modified Statistical Combination of Uncertainties," May, 1988.

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12.10 SECTION 10.0 REFERENCES

NONE

12.11 SECTION 11.0 REFERENCES

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