Appendix 4A. Tables

Table 4-1. Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters		Unit 1	Unit 2
1.	Reactor Core Heat Output, (100%), MWt	3469	3411
2.	Reactor Core Heat Output, 10^6 Btu/hr	11836.7	11648.8
3.	Heat Generated in Fuel, %	97.4	97.4
4.	System Pressure, Nominal, psia(1)	2280	2280
5.	System Pressure, Minimum Steady State, psia(1)	2250	2250
6.	Minimum DNBR at Nominal Conditions		
	Limiting Channel	2.9	2.9
7.	Minimum DNBR for Design Transients		
	Limiting Channel	(1) ≥1.55 (2) ≥1.50	1.45
8.	DNB Correlation	WRB-2M	WRB-2M
Core Flow(8}		**Unit 1**	**Unit 2**
9.	Total Thermal Flow Rate, 10^6 lbm/hr	145.5	144.8
10.	Effective Flow Rate for Heat Transfer, 10^6 lbm/hr	134.6	134.0
11.	Effective Flow Area for Heat Transfer, ft^2	51.1	51.1
12.	Average Velocity Along Fuel Rods, ft/sec	15.9	15.9
13.	Average Mass Velocity, 10^6 lbm/hr-ft^2	2.63	2.62

Coolant Temperature, °F (7)			
		Unit 1	**Unit 2**
14.	Nominal Inlet	552.0	554.7
15.	Average Rise in Vessel	63.2	62.5

16.	Average Rise in Core	67.8	66.4
17.	Average in Core	585.9	587.9
18.	Average in Vessel	585.1	587.5
Heat Transfer			
		Unit 1	**Unit 2**
19.	Active Heat Transfer, Surface Area, ft^2	59,866	59,866
20.	Average Heat Flux, Btu/hr-ft^2	192,579	189,360
21.	Maximum Heat Flux for Normal Operation, Btu/hr-ft^2	481,447	473,399
22.	Average Linear Power, kW/ft	5.53	5.44
23.	Peak Linear Power for Normal Operation, kW/ft[2]	13.8	13.58
24.	Peak Linear Power Resulting from Overpower Transients/Operator Errors (assuming a maximum overpower of 118%), kW/ft[3]	18.0	18
25.	Peak Linear Power for Prevention of Centerline Melt, kW/ft	>18.0	>18
26.	Power Density, kW per Liter of Core[(4)]	106.3	104.5
27.	Specific Power, kW per kg Uranium	39.4	38.8
Fuel Central Temperature			
			Robust Fuel Assembly
28.	Peak at Peak Linear Power for Prevention of Centerline Melt, °F		Burnup Dependent
29.	Pressure Drop[(5, 6)]		
	Across Core, psi		28.8 +/- 2.6
	Across Vessel, Including Nozzle psi		51.2 +/- 4.6
Items 30-64		Deleted duplicate information that is in Table 4-4. Moved entries that are not duplicative to Table 4-4. (i.e., Items 30, 33, 54, & 55)	

Notes:

1. Values used for thermal hydraulic core analysis.
2. This limit is associated with the value of F_Q = 2.50 and includes 2.6% gamma heating.
3. See Section 4.3.2.2.6
4. Based on cold dimensions
5. Based on best estimate reactor flow rate as discussed in Section 5.1.
6. RFA pressure drops are based on Reference 98 of Section 4.4.7.
7. These values are typical values. Values are based on RCS flow of 388,000 gpm and a bypass flow of 7.5%.
8. Based on a design flow of 388,000 gpm and nominal inlet temperatures.

Table 4-2. Analytical Techniques In Core Design

Analysis	Technique	Computer Code	Section Referenced
Fuel Rod Design			
Fuel Performance Characteristics (temperature, internal pressure,clad strain, etc.)	Semiempirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	PAD	4.2.3.1, 4.2.3.2 4.2.4.1 4.2.4.3
Nuclear Design			
1. Cross Sections and Group Constants	Microscopic data	Modified ENDF/B library	4.3.3
	Macroscopic constants for homogenized core regions	CASMO-3 or CASMO-4	4.3.3
	Group constants for control rods with self-shielding	CASMO-3 or CASMO-4	4.3.3
2. X-Y Power Distributions,Fuel Depletion, Critical Boron Concentrations, X-Y Xenon Distributions, Reactivity Coefficients	Diffusion Theory 3D, 2-Group Nodal Code	SIMULATE -3 or SIMULATE-3 MOX	4.3.3
3. Axial Power Distributions Control Rod Worths, and Axial Xenon Distribution	3D 2-Group Nodal Analysis Code	SIMULATE-3 or SIMULATE-3 MOX	4.3.3
4. Fuel Rod Power	Reconstructed Integral Rod Power	SIMULATE-3 or SIMULATE-3 MOX	4.3.4
5. Criticality of Reactor and Fuel Assemblies	2-D, Multi-group Transport Theory	CASMO-3	4.3.2.6
	3-D Monte Carlo	KENO-IV	

Analysis	Technique	Computer Code	Section Referenced
Thermal-Hydraulic Design			
1. Steady-state	Subchannel analysis of local fluid conditions in rod bundles, including the inertial and crossflow resistance terms	VIPRE-01	4.4.4.5
2. Transient Departure from Nucleate Boiling Analysis	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations	VIPRE-01	4.4.4.5.4

Table 4-3. Design Loading Conditions For Reactor Core Components

1.	Fuel Assembly Weight
2.	Fuel Assembly Spring Forces
3.	Internals Weight
4.	Control Rod Trip (equivalent static load)
5.	Differential Pressure
6.	Spring Preloads
7.	Coolant Flow Forces (static)
8.	Temperatures Gradients
9.	Differences in Thermal Expansion
	a. Due to temperature differences
	b. Due to expansion of different materials
10.	Interference Between Components
11.	Vibration (mechanically or hydraulically induced)
12.	One or More Loops Out of Service
13.	Operational Transients
14.	Pump Overspeed
15.	Seismic Loads (operation basis earthquake and safe shutdown earthquake)
16.	Blowdown Force (due to cold and hot leg break)

Table 4-4. Reactor Core Description

Active Core	**Robust Fuel Assembly**
Design	RCC Canless
Equivalent Diameter, in.	132.7
Core Average Active Fuel Height, in.	144.0
Height-to-Diameter Ratio	1.09
Total Cross-Section Area, ft^2	96.06
H_2 O/U Molecular Ratio, Lattice (68°F, 2250 psi)	~2.5
Reflector Thickness and Composition	
Top - Water plus Steel, in.	~10
Bottom - Water plus Steel, in.	~10
Side - Water plus Steel, in.	~15
Core Structure	
Core Barrell, ID/OD, in.	148.0/152.0
Thermal Shield	Neutron Pad Design
Fuel Assemblies	
Number	193
Rod Array	17 x 17
Rods per Assembly	264
Rod Pitch, in.	0.496
Overall Transverse Dimensions, in. (Typical)	8.426 x 8.426[(1)]
Fuel Weight (as UO_2), lbs.	220,012[(1)]
Zirconium Weight, lbs. (Cladding Surrounding Active Fuel)[(3)]	41,966[(1)]
	12
Composition of grids	INC718 Protective Grid, 2 INC718 End Grids, 6 ZIRLO Spacer Grids, 3 ZIRLO IFM Grids
Weight of Grids (Effective in Core) lbs.	INC-1066, ZIRLO-2820

Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	ZIRLO
Diameter of Guide Thimbles (upper part), in.	0.442 I.D. x 0.482 O.D.
Diameter of Guide Thimbles (lower part), in.	0.397 I.D. x 0.439 O.D.
Diameter of Instrument Guide Thimbles, in.	0.442 I.D. x 0.482 O.D.
Fuel Rods	
Number	50,592
Outside Diameter, in.	0.374
Diameter Gap, in.	0.0065
Clad Thickness, in.	0.0225
Clad Material	ZIRLO, Optimized ZIRLO
Fuel Pellets	
Material	UO_2 Sintered
Density (percent of Theoretical)	95.5
Fuel Enrichments w/o	0.711-5.0
Diameter, in.	0.3225
Length, in.	0.387 (chamfered) (enriched); 0.400-0.600 (chamfered) (axial blanket)
Mass of UO_2 per Foot of Fuel Rod, lb/ft	0.360 (1)
Hybrid Enhanced Performance Rod Cluster Control Assemblies (2)	
Neutron Absorber	B_4C
Diameter, in.	0.294
Density, lbs/in^3	0.064
Tip Material	Ag-In-Cd
Composition	80 percent, 15 percent, 5 percent (Ag-In-Cd)
Diameter, in.	0.301
Length, in.	40
Density, lbs/in^3	0.367 (Ag-In-Cd)

Cladding Material	Type 304 & 316, Cold Worked Stainless Steel
Clad Thickness, in.	0.0385
Number of Clusters	
Full Length	53
Number of Absorber Rods per Cluster	24
Full Length Assembly Weight (dry), lb.	94
Chrome Coated Next Generation Rod Cluster Control Assemblies	
Neutron Absorber	B_4C
Diameter, in.	0.294
Length, in.	102
Tip Material	Ag-In-Cd
Diameter, in.	
Lower Tip	0.296
Upper Tip	0.301
Length, in.	
Lower Tip	18
Upper Tip	22
Cladding Material	Type 304L Stainless Steel
Number of Full Length Clusters	53
Number of Absorber Rods per Cluster	24
Full Length Assembly Weight (dry), lb.	94
Hybrid Ionitrided Rod Cluster Assemblies	
Neutron Absorber	B_4C
Diameter, in.	0.294
Length, in.	102
Density, lbs/in^3	0.064
Tip Material	Ag-In-Cd
Composition	80 percent, 15 percent, 5 percent (Ag-In-Cd)
Diameter, in.	
Lower Tip	0.294
Upper Tip	0.300

Length, in.	
Lower Tip	12
Upper Tip	28
Density, lbs/in^3	0.367 (Ag-In-Cd)
Cladding Material	Type 316 Cold Worked Stainless Steel
Number of Full Length Clusters	
Unit 1	53
Unit 2	53
Number of Absorber Rods per Cluster	24
Full Length Assembly Weight (dry), lb.	94
Burnable Poison Rods	
Material	Al_2O_3-B_4C
Outside Diameter, in.	0.381
Clad Material	Zircaloy-4
Boron Loading	Proprietary
WABAs	
Material	Al_2O_3-B_4C
Inside Diameter, in.	0.225
Outside Diameter, in.	0.381
Clad Material	Zircaloy-4
Boron Loading	Proprietary

Notes:

1. Not exact for every core. Total weight will vary as region UO_2 varies. See region specific data for the most current values.
2. Information regarding the Westinghouse Hybrid EP-RCCAs has been retained for historical purposes. These RCCAs will be retained as potential spare RCCAs.
3. The values indicated are typical Mark-BW and RFA fuel assemblies.

Table 4-5. Nuclear Design Parameters

		Design Limits
Core Average Linear Power, kW/ft (based on 2.6% direct moderator heating)		Specified in Table 4-1, Item 22
Total Heat Flux Hot Channel Factor, F_Q		Specified in the COLR
Reactivity Coefficients		
Doppler-only Power Coefficients, pcm/% power		
(See Figure 15-3)	Upper Curve	-19.4 to -12.6
	Lower Curve	-9.5 to -6.0
Fuel Temperature Coefficient, pcm/°F	(BOL)	≤ -0.9
	(EOL)	≤ -1.2
Moderator Temperature Coefficient, pcm/°F		
Most pos BOL (0-70% FP)		≤ 7.0
Most pos BOL (70-100% FP)		≤ -0.233
Most pos EOL HFP		≤ -24
Most pos EOL HZP		≤ -10
Most neg EOL HFP		> -51
Boron Coefficient, pcm/ppm		≤ -5
Delayed Neutron Fraction and Lifetime		
l BOL - (min) μsec		> 16
l BOL - (max) μsec		< 22
l EOL - (min) μsec		≥ 18
l BOL - (max) μsec		< 32
β_{eff} BOL - (min)		> 0.0055
β_{eff} BOL - (max)		< 0.0070
β_{eff} EOL - (min)		> 0.0040
β_{eff} EOL - (max)		< 0.0060
Control Rods		
Rod Worths	See Table 4-7	
Maximum Bank Worth, pcm	See Chapter 15	
Maximum Ejected Rod Worth	See Chapter 15	

Note:

1. 1 pcm = (percent mille rho) = 10^{-5} $\Delta\rho$ where $\Delta\rho$ is calculated from two statepoint values of K_{eff} by ln (K_2/K_1)

Table 4-6. Nuclear Design Parameters. HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Boron Concentrations (ppm) (First Cycle)	
Zero Power, K_{eff} = 1.00, Cold, Rod Cluster	
Control Assemblies Out, 1 percent uncertainty included	1650
Zero Power, K_{eff} = 1.00, Hot, Rod Cluster	
Control Assemblies Out, 1 percent uncertainty included	1500
Design Basis Refueling Boron Concentration	2000
Zero Power, K_{eff} = 1.00, Cold, Rod Cluster	
Control Assemblies In, 1 percent uncertainty included	1000
Zero Power, K_{eff} = 1.00, Hot, Rod Cluster	
Control Assemblies Out	1400
Full Power, No Xenon, K_{eff} = 1.0, Hot, Rod	
Cluster Control Assemblies Out	1350
Full Power, Equilibrium Xenon, K_{eff} = 1.0,	
Hot Rod Cluster Control Assemblies Out	1050
Reduction with Fuel Burnup	
First Cycle, ppm/GWD/MTU[1]	See Figure 4-20.
Reload Cycle, ppm/GWD/MTU	~100

Note:

1. Gigawatt Day (GWD) = 1000 Megawatt Day (1000 MWD). During the first cycle, fixed burnable poison rods are present which significantly reduce the boron depletion rate compared to reload cycles.

Table 4-7. Reactivity Requirements For Rod Cluster Control Assemblies. HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

Reactivity Effects, percent	**Beginning of Life (First Cycle)**	**End of Life (First Cycle)**	**End of Life (Equilibrium Cycle) (Preliminary)**
1. Control requirements			
Fuel temperature (Doppler), percent $\Delta\rho$	1.28	1.10	1.10
Moderator temperature, percent $\Delta\rho$	.10	0.80	1.10
Void, percent $\Delta\rho$	.01	.05	.05
Redistribution, percent $\Delta\rho$	.50	.85	.95
Rod Insertion Allowance, percent $\Delta\rho$	.50	.50	.50
2. Total Control, percent $\Delta\rho$	2.39	3.30	3.70
3. Estimated Hybrid Rod Cluster Control Assembly Worth (53 Rods)			
a. All full length assemblies inserted, percent $\Delta\rho$	8.53	8.03	7.65
b. All but one (highest worth) assemblies inserted, percent $\Delta\rho$	7.23	6.90	6.49
4. Estimated Rod Cluster Control Assembly credit with 10 percent adjustment to accommodate uncertainties (3b - 10 percent), percent $\Delta\rho$	6.51	6.21	5.84
5. Shutdown margin available (4-2), percent $\Delta\rho$	4.12	2.91	2.14[1]

Note:

1. The design basis minimum shutdown is 1.3%$\Delta\rho$

Table 4-8. UO_2 Benchmark Critical Experiments

Deleted Per 2007 Update.

UO_2 Critical Experiments for SCALE 4.4 Methodology

No.	Ref.	General Description	Enrichment W% U^{235}	Poison Material	Poison Thickness (cm)	Critical Separation (CM) X	Critical Separation (CM) Y	Critical No. of Rods
51	60	Multiple Fuel Clusters	4.31	None	-	4.72	4.72	253.8
53	60	Multiple Fuel Clusters	4.31	None	-	6.61	6.61	432.7
55	60	Multiple Fuel Clusters	4.31	None	-	2.83	14.98	396
56	60	Multiple Fuel Clusters	4.31	None	-	2.83	19.81	432
57	60	Multiple Fuel Clusters	4.31	None	-	2.83	13.64	360
58	60	Multiple Fuel Clusters	4.31	None	-	2.83	12.02	288
59	60	Multiple Fuel Clusters	4.31	None	-	2.83	11.29	252
60	60	Multiple Fuel Clusters	4.31	None	-	2.83	10.86	234
61	60	Multiple Fuel Clusters	4.31	None	-	2.83	8.38	225
62	60	Multiple Fuel Clusters	4.31	None	-	2.83	0	219.2

No.	Ref.	General Description	Enrichment W% U^{235}	Poison Material	Poison Thickness (cm)	Critical Separation (CM) X	Critical Separation (CM) Y	Critical No. of Rods
64	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	2.83	247.1
65	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	4.54	270
66	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	3.38	252
67	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	6.49	342
68	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	9.96	432

No.	Ref.	General Description	Enrichment W% U^{235}	Poison Material	Poison Thickness (cm)	Critical Separation (CM) X	Critical Separation (CM) Y	Critical No. of Rods
69	60	Multiple Fuel Clusters	4.31	SS-304	.302	2.83	11.55	450
6D	60	Multiple Fuel Clusters	4.31	None	-	2.83	2.83	221.3
70	60	Mutiple Fuel Clusters	4.31	SS-304	.302	2.83	8.10	396
71	60	Multiple Fuel Clusters	4.31	SS-304	.485	2.83	2.83	271.8
72	60	Multiple Fuel Clusters	4.31	SS-304	.485	2.83	4.47	306
73	60	Multiple Fuel Clusters	4.31	SS-304	.485	2.83	8.36	432
83	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	2.83	642.5
84	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	6.61	669.8
85	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	8.5	675.9
94	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	8.5	663.3
95	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	4.72	633.5
96	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	3.6	616
97	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	2.83	601
98	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	2.83	597.9
100	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	4.72	631.2
101	60	Multiple Fuel Clusters	4.31	Boraflex	.226	2.83	6.61	650.8
105	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	2.83	643.1
106	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	4.94	660
107	60	Multiple Fuel Clusters	4.31	Boraflex	.452	2.83	6.61	672.2
131	60	Multiple Fuel Clusters	4.31	None	-	12.27	N/A	3-12x16

No.	Ref.	General Description	Enrichment W% U^{235}	Non-Fuel Pins	Pin Lattice Spacing (cm)	Lattice Width (rods)	Critical No. of Rods
43	60	Single Lattice	4.31	None	1.892	17	218.6
45	60	Single Lattice	4.31	None	1.892	14	216.2
46	60	Single Lattice	4.31	None	1.892	12	225.8
47	60	Single Lattice	4.31	25 water holes	1.892	14	167.6
48	60	Single Lattice	4.31	25 al clad voids	1.892	14	203.0
4C	60	Single Lattice	4.31	None	1.892	18	223.0
96	60	Single Lattice	2.35	None	1.684	23	523.9
97	60	Single Lattice	2.35	25 water holes	1.684	23	485.8

No.	Ref.	General Description	Enrichment W% U^{235}	Poison Material	Distance from SS plate to Fuel Cluster(cm)	Length by Width of Array	Critical Spacing Between Clusters (cm)
14	61	3 x 1 Arrays	2.35	None	-	20 x 16	8.42
15	61	3 x 1 Arrays	2.35	None	-	20 x 17	11.92
21	61	3 x 1 Arrays	2.35	None	-	20 x 14	4.46

No.	Ref.	General Description	Enrichment W% U^{235}	Poison Material	Poison Thickness	Distance from SS plate to Fuel Cluster (cm)	Length by Width of Array	Critical Spacing Between Clusters (cm)
26	61	3 x 1 Arrays	2.35	SS-304	0.302	4.04	20 x 16	7.76
27	61	3 x 1 Arrays	2.35	SS-304	0.302	0.64	20 x 16	7.42
34	61	3 x 1 Arrays	2.35	SS-304	0.302	0.64	20 x 17	10.44
35	61	3 x 1 Arrays	2.35	SS-304	0.302	4.04	20 x 17	11.47
5	61	3 x 1 Arrays	2.35	SS-304	0.485	2.73	20 x 16	7.64

28	61	3 x 1 Arrays	2.35	SS-304	0.485	0.64	20 x 16	6.88
29	61	3 x 1 Arrays	2.35	SS-304	0.485	4.04	20 x 16	7.51

No.	Ref.	General Description	Enrichment W% U^{235}	Boral Poison Loading (g B/cm^2)	Flux Trap Width (cm)	Flux Trap to Fuel Separation (CM)		Critical No. of Rods
						X	Y	
214	62	Neutron Flux Traps	4.31	0.36	3.73	0.295	0.295	952
223	62	Neutron Flux Traps	4.31	0.36	3.73	4.077	4.077	858
224	62	Neutron Flux Traps	4.31	0.36	3.73	2.186	2.186	874
229	62	Neutron Flux Traps	4.31	0	3.81	0.295	0.295	308
230	62	Neutron Flux Traps	4.31	0.05	3.75	0.295	0.295	855

Note:

1. Percentages refer to weight percent boron content

Table 4-9. Axial Stability Index Pressurized Water Reactor Core With A 12 Foot Height

Burnup (MWD/MTU)	F_Z	C_B (ppm)	Stability Index (hr^{-1})	
			Exp	Calc
1550	1.34	1065	-0.041	-0.032
7700	1.27	700	-0.014	-0.006
		Difference:	+0.027	+0.026

Table 4-10. Typical Neutron Flux Levels (n/cm^2-sec) At Full Power

	E > 1.0 Mev	**5.53 Kev <E ≤ 1.0 Mev**	**6.25 ev ≤E <5.53 Kev**	**E < .625 ev $(nv)_0$**
CORE CENTER	6.51×10^{13}	1.12×10^{14}	8.50×10^{13}	3.00×10^{13}
CORE OUTER RADIUS AT MIDHEIGHT	3.23×10^{13}	5.74×10^{13}	4.63×10^{13}	8.60×10^{12}
CORE TOP, ON AXIS	1.53×10^{13}	2.42×10^{13}	2.10×10^{13}	1.63×10^{13}
CORE BOTTOM, ON AXIS	2.36×10^{13}	3.94×10^{13}	3.50×10^{13}	1.46×10^{13}
PRESSURE VESSEL INNER WALL, AZIMUTHAL PEAK, CORE MIDHEIGHT	2.77×10^{10}	5.75×10^{10}	6.03×10^{10}	8.38×10^{10}

Table 4-11. Deleted Per 1998 Update

Table 4-12. Deleted Per 2001 Update

Table 4-13. Saxton Core II Isotopics Rod My, Axial Zone 6

Atom Ratio	Measured[1]	2σ Precision (%)	Leopard Calculation
U-234/U	4.65×10^{-5}	±29	4.60×10^{-5}
U-235/U	5.74×10^{-3}	±0.9	5.73×10^{-3}
U-236/U	3.55×10^{-4}	±5.6	3.74×10^{-4}
U-238/U	0.99386	±0.01	0.99385
Pu-238/Pu	1.32×10^{-3}	±2.3	1.222×10^{-3}
Pu-239/Pu	0.73971	±0.03	0.74497
Pu-240/Pu	0.19302	±0.2	0.19102
Pu-241/Pu	6.014×10^{-2}	±0.3	5.74×10^{-2}
Pu-242/Pu	5.81×10^{-3}	±0.9	5.38×10^{-3}
Pu/U[2]	5.938×10^{-2}	±0.7	5.970×10^{-2}
Np-237/U-238	1.14×10^{-4}	±15	0.86×10^{-4}
Am-241/Pu-239	1.23×10^{-2}	±15	1.08×10^{-2}
Cm-242/Pu-239	1.05×10^{-4}	±10	1.11×10^{-4}
Cm-244/PU-239	1.09×10^{-4}	±20	0.98×10^{-4}

Notes:

1. Reported in Reference 29
2. Weight ratio

Table 4-14. Critical Boron Concentrations, HZP, BOL

Plant Type	Measured	Calculated
2-Loop, 121 Assemblies 10 foot core, ppm	1583	1589
2-Loop, 121 Assemblies 12 foot core, ppm	1625	1624
2-Loop, 121 Assemblies 12 foot core, ppm	1517	1517
3-Loop, 157 Assemblies 12 foot core, ppm	1169	1161
3-Loop, 157 Assemblies 12 foot core, ppm	1344	1319
4-Loop, 193 Assemblies 12 foot core, ppm	1370	1355
4-Loop, 193 Assemblies 12 foot core, ppm	1321	1306

Table 4-15. Benchmark Critical Experiments B_4C Control Rod Worth

WREC Critical Experiment	No. Of Fuel Rods	No. Of Control Rods	Measured[1] Worth, %Δρ	Calculated Worth %Δρ
2A	888	12 .395″ O.D. B_4C	8.20	8.37
3B	888	12 .232″ O.D. B_4C	4.81	4.82
4B	884	16 .232″ O.D. B_4C	6.57	6.35
5B	945	16 .232″ O.D. B_4C	5.98	5.83

AG-IN-CD Comparison of Measured and Calculated Rod Worth

4-Loop Plant, 193 Assemblies, 12-foot core	Measured (pcm)	Calculated (pcm)
Bank D	1403	1366
Bank C	1196	1154
All Rods In Less One	6437	6460
ESADA Critical[2], 0.69 Inch Pitch, 2 w% PuO_2, 8% Pu^{240} 9 Control Rods		
6.21 inch rod separation	2250	2250
2.07 inch rod separation	4220	4160
1.38 inch rod separation	4100	4019
Line Item Deleted Per 2001 Update		

Note:

1. The measured worth was derived from the calculated value of ln k_1/k_2, where k_1 and k_2 were calculated with the measured buckling before and after insertion of the control rods, which replace fuel rods in arrays at the center of the experiment. The standard deviation in the measured worth is about 0.3% $\Delta\rho$ based on the uncertainties in the measured axial buckling.

2. Reported in Reference 30.

Table 4-16. Comparison Of Measured And Calculated Moderator Coefficients At HZP, BOL

Plant Type/ Control Bank Configuration	Measured α_{iso}(1) (pcm/°F)	Calculated α_{iso}(2) (pcm/°F)
3-Loop, 157 Assemblies, 12 foot core		
D at 160 steps	-0.50	-0.50
D in, C at 190 steps	-3.01	-2.75
D in, C at 28 steps	-7.67	-7.02
B, C, and D in	-5.16	-4.45
2-Loop, 121 Assemblies, 12 foot core		
D at 180 steps	+0.85	+1.02
D in, C at 180 steps	-2.40	-1.90
C and D in, B at 165 steps	-4.40	-5.58
B, C, and D in A at 174 steps	-8.70	-8.12
4-loop, 193 assemblies, 12 foot core		
ARO	-0.52	-1.2
D in	-4.35	-5.7
D + C in	-8.59	-10.0
D + C + B in	-10.14	-10.55
D + C + B + A in	-14.63	-14.45

Notes:

1. Isothermal coefficients, which include the Doppler effect in the fuel.
2. $\alpha_{iso} = 10^5 \ln \frac{k_2}{k_1} \ / \Delta T°F$

Table 4-17. Deleted Per 2000 Update

Table 4-18. Deleted Per 1993 Update

Table 4-19. Void Fractions At Nominal Reactor Conditions With Design Hot Channel Factors

HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED

	Average	***Maximum***
Core	*0.0*	—
Hot Subchannel	*1.5*	*3.5*

Table 4-20. Measurements Required In The Calculation Of Reactor Flow Using A Calorimetric Technique

Parameter	Instrument	Function
1. Feedwater venturi pressure differential	Rosemount ΔP gauge and compatible readout	feedwater flow[1]
2. Feedwater temperature	Continuous lead thermocouple	feedwater enthalpy and density[1]
		venturi thermal expansion
3. Steam pressure	Transducer and process computer readout	steam enthalpy
4. Reactor coolant T_{hot}	Narrow range RTD and data acquisition system or DVM readout	RCS hot leg enthalpy
5. Reactor coolant T_{cold}	Narrow range RTD and data acquisition system or DVM readout	RCS cold leg enthalpy RCS specific volume
6. Reactor coolant pressure	Transducer and process computer readout	RCS enthalpy and specific volume
Other information required for the calculation is as follows:		
7. Feedwater venturi coefficient from vendor calibration.		
8. Primary system heat losses and pump heat input obtained from calculations.		

Notes:

1. In addition to the originally-installed venturi flow nozzle instruments, ultrasonic flow meters were later installed on Unit 1 to provide more precise feedwater measurement. These ultrasonic flowmeters measure both feedwater flow and temperature, and provide input to the core power calorimetric calculation.

Table 4-21. Statistically Combined Uncertainty Factors for Fq, FΔH, and Fz

Uncertainty Factor	MODEL	Uncertainty Factor Value
Fq-SCUF	CASMO-3/SIMULATE-3P	1.071
FΔH-SCUF	CASMO-3/SIMULATE-3P	1.040
Fz-SCUF	CASMO-3/SIMULATE-3P	1.053
Low Enriched Uranium (LEU) Fuel		
Fq-SCUF	CASMO-4/SIMULATE-3 MOX	1.0735
FΔH-SCUF	CASMO-4/SIMULATE-3 MOX	1.04 (SCD) 1.032 (Non-SCD) (2)
Fz-SCUF	CASMO-4/SIMULATE-3 MOX	1.049
Mixed Oxide (MOX) Fuel		
Fq-SCUF	CASMO-4/SIMULATE-3 MOX	1.078
FΔH–SCUF	CASMO-4/SIMULATE-3 MOX	1.04 (SCD) 1.035 (Non-SCD) (2)
Fz-SCUF	CASMO-4/SIMULATE-3 MOX	1.049

Note:

1. The CASMO-4/SIMULATE-3 MOX uncertainties are based on values in DPC-NE-1005-P-A, the values shown above have been increased to ensure that they remain bounding.
2. Non-SCD FΔH–SCUF excludes engineering hot channel factor uncertainty.

Table 4-22. Elbow Tap Coefficients

	Unit 1	Unit 2
Loop A Tap I	0.29773	0.30680
Loop A Tap II	0.29348	0.29606
Loop A Tap III	0.29515	0.30382
Loop B Tap I	0.30410	0.30313
Loop B Tap II	0.30803	0.28601
Loop B Tap III	0.30444	0.30689
Loop C Tap I	0.28915	0.31712
Loop C Tap II	0.28489	0.29659
Loop C Tap III	0.29097	0.30389
Loop D Tap I	0.30331	0.29936
Loop D Tap II	0.29932	0.29929
Loop D Tap III	0.31051	0.30137

Note:

Do not delete table. Elbow tap coefficients are committed to be included in UFSAR by Duke Letter to the NRC dated February 26, 2003 and NRC Issuance of Amendment 199 dated March 19, 2003.

Table 4-23. Fuel Assembly Design Information for Current Demonstration Programs

Parameter	NGF [1]	MOX [1]
Total Number of Assemblies in Test Program	8	4
Overall Transverse Dimensions, in. (Typical)	8.434	8.437
Rod Cladding Material	Optimized ZIRLO ™	M5 ™
Rod Length, in.	152.80	152.40
Rod Outside Diameter, in.	0.3740	0.3740
Rod Pitch, in.	0.496	0.496
Fuel Density (percent of Theoretical	95.5	95.0
Fuel Pellet Material	UO_2	MOX
Fuel Weight (as UO_2/MOX), lbs.	1139	1157 [2]
Composition of Guide Thimbles	Optimized ZIRLO ™	M5 ™

Notes:

All values are typical or reference values for the design.

Includes plutonium and uranium dioxide.

Table 4-24. Mechanical and Thermal Hydraulic Analysis Methods for Current Demonstration Programs

NGF Demonstration Program

The NGF assemblies are analyzed with the same methods as those contained in UFSAR Section 4.2.3 and 4.4.1.

MOX Demonstration Program

BAW-10231P-A, Rev. 1, COPERNIC Fuel Rod Design Computer Code, January 2004.

DPC-NE-2005P-A, Rev. 3, Thermal-Hydraulic Statistical Core Design Methodology, September 2002.